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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

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November 19, 1959

To: M. E. Ramsey, C. J. Borkowski, E. P. Epler, F. W. Manning, R. H. Ritchie, C. E. Winters, E. O. Wollan, F. Kertesz

Subject: Information for Review of ORR Operation

The ORR has operated for approximately a year and a half, and operations are becoming reasonably routine. Corrosion, which appeared dangerous last year, has been satisfactorily controlled except in a few cases; and troubles in the controls have been partially solved. Improved design standards have reduced trouble from experiment instrumentation, but serious problems exist in maintenance. A great many experiments have been installed, but an almost equal number await installation. Dangerous effects of beam holes on reactor controls have been solved. Pool experiments which may have radioactive lines are one of the current problems. Many experimenters wish to drain the pool to do installation work, and this cannot be done if the piping of other experiments is very radioactive.

The increase in power of the ORR to 30 Mw appears to be feasible, but a number of problems remain to be solved before the power is raised beyond this. These are chiefly concerned with increased pressure and thermal stresses in the tank.

Experiments

Pool Experiments

It appears that we will have to consider again the decision, made about a year ago, that all experiments be designed so that they can be installed and removed without lowering the pool water. This decision was made because it seemed likely that many experiments would release fission products inside piping in the pool. Recently some experimenters have questioned this decision on the grounds that it may be more expensive to install and remove experiments under water than to install them with the water drained and provide shielding around the radioactive pipes. If the worst shielding conditions were considered, as much as eight inches of lead may be required for some experiments; and there is a question as to whether such a weight could be supported in the pool or if enough room is available. The alternatives appear to be as follows:

1. To re-affirm the previous decision to require all experiments to be designed so that they can be installed and removed without lowering the

pool water and without excessive down time. (This may result in certain experiments costing much more for design and construction than would be the case if they were installed with the water lowered.)

2. The specifications for pool experiments could be changed, requiring that all such experiments be shielded in place so that the pool water can be drained to the top of the reactor tank. (It is not certain that this is feasible, at least in the case of the GCR gas-cooled loop, since such large amounts of shielding would be required.)

3. Specifications for pool experiments could be changed so that no experiments which might have radioactive lines be permitted in the pool, thus eliminating the need for shielding at least if much is required. (This would probably eliminate the GCR loop.)

4. All experiments in the pool could be required to install good monitoring equipment so that the reactor could be shut down immediately if a release of radioactivity <u>within</u> the experiment system occurred. If the experiments became only slightly radioactive (up to 10 R/hr), temporary shielding could be used, permitting the water level to be lowered. If the radiation was still too high, the water level would not be lowered until a later cycle. During the intervening time the "hot" experiment would be cleaned as well as possible by flushing with decontaminating solutions. If it was still too radioactive after this treatment, a decision would have to be made as to whether the hot experiment should be taken out or the water level kept high for the remainder of the experiment's life. This would probably mean that other experiment work in the pool would be delayed.

Beam-Hole Experiments

The beam-hole plugs have all been modified to permit filling and draining of the plugs while the reactor is operating. Since an aluminum line in a concrete beam-hole plug has corroded to the extent of plugging the line, it has been decided that the beam-hole liners should be kept drained and dry. (The collimator plugs may be filled and drained.)

All beam-hole shutters have been installed and five of the six beam-hole shields have been set in place. Four of the experiments are now in operation and the external equipment for the other two is being installed. Approval for construction of a beam-hole enclosure has been requested of the Policy Committee. The enclosure will consist of metal panels ten feet high around the beam-hole area. Acoustical tile will be placed on the east wall of the building up to 30 feet high.

In-Pile Experiments

Five major experiments utilizing the access flanges through the tank top have operated successfully. Two in-pile experiments are being reviewed by the Operations Division and another is expected to be submitted for review within the next month.

A homogeneous fuel loop was operated in HN-1 facility from May until September. No unusual incidents were encountered as a result of the operation of the loop.

A proposal to install a contaminated loop in the south facility, formerly assigned to the PAR experiment, is pending approval. This loop would require a shielded, gas-tight, equipment room about 40 feet in length along the south side of the reactor. Also a new access door would have to be made in the south side of the reactor building through which the loop would be brought into, and removed from, the building on a track.

It has been proposed that a new laboratory, similar to the cryostat laboratory on the south side of the center pool, be constructed on the north side where there is a shortage of space for experiment equipment near the pool. The south balcony is very crowded already and there is still much equipment to be installed in this area. Some of the experimenters are moving to the north side of the reactor since there is more room available on that side. An equipment cubicle will be built on the north balcony to accommodate the compressors for the GCR loop.

Reactor Controls

Instrument Maintenance

An urgent need exists for a qualified engineer to work on maintenance of reactor and experiment instrumentation. Under the present system, very little preventive maintenance can be done. Only after an instrument fails is it serviced. An example of this sort of trouble occurred on the week end of November 14-15 following a fuel reloading. The reactor was started up, but one of the control rods dropped several times within a few hours. The reactor was shut down, the control-rod drive partially dismantled, and the magnet was found to be damp apparently from water which had leaked around the drive-tube seals. This was replaced after about eight hours and an attempt made to start the reactor. It was then found that the No. 3 power safety channel was defective and would not scram the rods when tested. The amplifier and the preamp were checked and found to be functioning properly, and it was then found that the chamber was defective. Since changing the chamber would involve a shutdown of several days, it was necessary to operate with two safety channels. At about the same time or shortly thereafter, the following instruments failed: N^{16} activity, pH of pool secondary system, and ΔT on reactor. The power recorder has never functioned properly. A spare safety chamber, installed in the reactor expressly for such a condition where one of the other chambers failed, was found to be defective. If the ORR is to achieve good "on stream" time, it is absolutely necessary that we improve our preventive maintenance. We have already made great improvements in the experiment instrumentation by standardizing design and by using tested and proved components. However, little has been done in the way of preventive maintenance. Other reactors such as those at Harwell have had expensive instruments added simply to avoid shutdowns through instrument failure. For example, the PLUTO reactor has been converted to a coincidence, or two-out-of-three, system whereby two out of three safety channels must indicate unsafe conditions before the reactor will scram. This eliminates shutdowns caused by failure of one channel. So far, however, at ORNL we have not adopted such a system but depend upon a one-out-of-two or a one-out-of-three system whereby any one channel may actuate a scram. Unless we can improve our maintenance it would seem that we may have to adopt a coincidence system.

Miscellaneous Changes

Additional modifications to the electronics and minor adjustments on the mechanical controls have reduced the spurious scrams markedly. Modifications include: (1) installing new and stronger magnets, (2) installing dual magnet amplifiers in parallel for each magnet; and (3) adjusting the air gap between keeper and magnet to 0.167 inches (the correct calculated value) when the rod is released. The combination of these changes has reduced the shutdowns caused by "unknown reasons" to one since the July shutdown.

A major improvement in the servo system permits "controlled automatic shimming" during start-up. When taking the reactor from ~1% power to full power, a shim rod is selected as "preferred" and the servo demand is raised. The servo system automatically raises the power by withdrawing the servo-controlled rod until the withdraw limit switch is energized. At this time, the preferred rod is withdrawn until the power exceeds the demand and clears the withdraw-limit switch by inserting the servo-controlled rod. This cycle will continue until the desired level has been obtained.

Gamma chambers have been installed in the pool adjacent to the reactor vessel to safeguard the reactor against the effects of draining and filling the beam holes. The gamma chambers are not affected by the beam holes, while the neutron chambers may vary by a factor of five. The northeast and southeast corners of the vessel were selected for gamma chamber locations. These are connected to the reactor scram circuit via recorder switches. They have not been attached to the sigma safety system (fast scram). However, their main purpose is to guard against the effects of inadvertent draining or filling of the beam-hole liners on the reactor control system; and this does not require very fast response.

Shutdown Experience

Improvement in experiment instrumentation resulted in a markedly reduced number of shutdowns compared to the LITR. Only approximately five occurred in the first nine months of 1959 as shown in the following table.

ANALYSIS OF UNSCHEDULED SHUTDOWNS
JANUARY - SEPTEMBER 1959

	Number		Hours	
Instrument failure		24		24.729
Operations	19		23.629	
Experiments	5		1.100	
Human Error		3		7.017
Operations	2			
Experiments	1			
Power failure		5		12.950
TOTAL		32		44.696

CAUSES OF SHUTDOWNS DUE TO EXPERIMENTS IN ORR
FIRST NINE MONTHS 1959

Date	Experiment	Cause
6-12-59	HN-1	Faulty servo follower gave scram.
7-17-59	HN-1	Setback due to high loop pressure indication on "Rotex" instrument. This was not indicated on another pressure instrument. Faulty instrument.
7-28-59	HN-1	Failure on "Servo Follower" instrument* gave scram. Other instruments did not indicate abnormal condition.
8-5-59	GE (F-2)	Loss of power to instruments caused scram.
9-12-59	GE	Defective thermocouple on recorder with upscale burnout gave setback.

*This instrument is a slave to the pressurizer-temperature controller. After this incident the following changes were made: (a) eliminated the setback, (b) transferred the scram to the master, (c) alarm was left on slave.

RELEASE AND DROP TIMES OF ORR CONTROL RODS

Time in milliseconds; water flow, 16,000 gpm
Rods dropped from 20 in. except where noted

	Rod No.	5-21-59	6-20-59	7-17-59	8-14-59	9-11-59	10-9-59	11-3-59
Release time	3	17.5 & 18	19	7.9 & 6.8	16	9.3	6.8	9.4
Time of flight	3	170	110	205	265	220	240	245*
Release time	4	14 & 14	14.5	11.8 & 10.4	23.5	15	14	19.5
Time of flight	4	140	310	285	290	180	270	250*
Release time	5	13 & 12	14.5	7.9 & 8.1	17.5	11	11.8	18
Time of flight	5	90	110	275	290	210	300	270*
Release time	6	19 & 18	21	9.5 & 9.0	18	9	7.6	11
Time of flight	6	240	280	275	275	180	280	240

*Noted as being from 10 in. with 16,200 gpm flow.

New Shim Rods

The present four control rods do not provide enough control to permit operation of the reactor through a complete cycle. It is proposed to provide two additional Be-Cd rods in the reflector on the east side of the lattice. In operation, these would probably be pulled out first so that the effect on the experiments nearby would be minimized. Some investigation is also underway on the feasibility of using Cd-Al in place of the Cd-fuel rods. If this should prove to be feasible the operating costs would be reduced, and the effect of the rods would be much more stable. The effect of the present Cd-fuel rods varies considerably as the fuel burns out.

Radiation Incidents

Five incidents have occurred in the ORR which increased the normal background appreciably.

1. In March while transferring an irradiated sample (throium oxide-uranium oxide) from the reactor vessel to the pool, radioactive gas escaped and required the evacuation of the building for about one hour.

2. A loose pipe connection on a flow measuring unit sprayed several gallons per minute of reactor water into the pipe chase. At the time of the incident an off-gas sweep was not provided for the pipe chase, and radioactive gases escaped into the building proper. A temporary off-gas line attached to the pipe chase area reduced the building activity to a tolerable level which permitted the continued operation of the reactor until the next shutdown when it was possible to repair the damaged line.

3. A seal leak on a fuel experiment installed in core position B-9 on 4-23-59 developed shortly after the reactor reached full power. Air activity in the building began to increase, the reactor was shut down, and the sample pulled above the flux zone. This eliminated the air activity problem and permitted the reactor to be started up.

4. Argon^{41} activity was dispersed throughout the building via the warm drain system in September. This occurred while checking the HB-3 liner to determine its status by making a slight air purge to the off-gas system. The activated gases apparently escaped into the building through an improperly seated valve on the warm drain system. The building was evacuated for about 20 minutes.

5. On May 5, 200 mg of iodine was irradiated in the ORR hydraulic tube for 30 minutes. The material was encapsulated in a nylon holder which ruptured. Air activity in the vicinity of the loading-unloading station (where the sample stops after removal from the reactor) increased from 500 c/m to about 1500 c/m on a CAM when the loading station was opened. The general background was 80 mr/hr (cp with paper shell chamber) over the west pool.

An analysis of pool water indicated I^{128} activity but the intensity was low. Smears on the floor area and pool parapet were less than 50 c/m.

Corrosion

The vacuum system which was installed to dry the embedded piping of the reactor system has been operating effectively. Periodic checks of condensate indicate no appreciable change since the lines were first dried. The system thus also serves as a leak detector.

A corrosion test of aluminum under different ORR conditions has been in progress for about a year. Inspections have been made at about six-month intervals. The second inspection revealed various degrees of corrosion and pitting with aluminum pipe surrounded by wetted glass wool showing the worst condition. The program is continuing. Neither glass wool nor concrete cause corrosion under dry conditions.

A failure in the water supply piping embedded in a beam-hole plug was observed on September 9. Disassembly revealed pitting which penetrated the wall of the tube. This plug was one of the six original beam-hole plugs and had been in service for about 18 months. The remaining five plugs, along with a special new plug in HB-6, are presently in service. There is no assurance that the same condition will not occur in these plugs, and special precautions will be taken to prevent water getting in the liner section. The ruptured tube is under study by Metallurgy to determine the cause.

Continuous monitoring of the ORR water systems indicates that corrosion rates (of the inside surfaces) are insignificant. Corrosion coupons regularly tested indicate rates in the order of .1 mpy or less.

A second inspection of the shell-tube heat exchanger was made during the July shutdown. The tube bundle and shell appear to still be in good condition with no evidence of significant corrosion. A considerable amount of silt had collected inside the shell portion of the unit. A routine flushing of the unit is now being done to eliminate this condition. Because of the possible long downtime required if the heat-exchanger bundle should fail, a spare bundle will be purchased as soon as Policy Committee approval is obtained.

Na^{24} has recently been found in some of the water pumped from sumps along side the buried reactor cooling piping. This indicates that some small leaks have developed in the buried piping. It appears that the amount of leakage from the system is quite small and there appears to be a good possibility that once a leak is established the demineralized water will wash the surface of the hole and prevent further corrosion. The buried Al pipe was originally tarred and wrapped, and it is believed that corrosion will occur only where the protective coating has been scratched through to the aluminum. The sumps are monitored regularly for Na^{24} activity and the leakage from the system is checked so that if the rate increases to an appredíable value it will be detected. The sump pumps were installed to keep the ground water level below the line, many portions of which were often below water.

Water System

Equilibrium radioactivities in the ORR water systems remain about the same. The ion exchange units maintain these levels at about 25,000 c/m/ml in the reactor system and 600 c/m/ml in the pool. The degasifier has been in service since March, and its effectiveness is measured by a reduction in general background by a factor of 3. Since installation, the occasions where fission gases are released from the water (particularly during reloading) have been practically eliminated.

Trane Coolers

In March the modifications on the air coolers were completed, but performance tests indicated no significant improvement in heat transfer. This deficiency in cooling limited the operating level during the hot summer months to 16 Mw. With the cooler weather the reactor is again operating at 20 Mw when the ambient temperature does not exceed 82°F.

Power Level Increase

Little trouble is anticipated in raising the power to 30 Mw once the additional cooling and emergency shutdown cooling are installed. It is hoped that this can be done by April 1960, but delays in obtaining approval have set back the schedule and it now seems likely that it will be June before the additional cooling is ready. After April it will be necessary to reduce power to ~16 Mw until the new coolers are ready (unless it can be proved that the temperature of the reactor structure can be safely raised). Following operation at 30 Mw, it is planned to move as rapidly as possible to increase the power as high as is deemed feasible. It is hoped that a power level of 45 Mw can be safely achieved.

Increasing the power level of the ORR above 30 Mw involves mainly optimization of the following: (a) increased water flow; (b) operation of the fuel plates at higher temperatures; (c) reduction of the temperature of hot spots which limit the power; (d) insurance against damage to the reactor tank or other structure by increased stress or heating; and (e) provision for emergency pumping capacity to cool the fuel in event of a power failure. These problems are under study and are treated in more detail in the appendix.

The problems which appear to be most difficult are those relating to the increased tank pressure due to the need for pumping more water through the reactor and the thermal stresses encountered at higher power. These problems are being studied by a group in the Reactor Projects Division, and a memorandum from M. W. Rosenthall is attached in the appendix.

Temperatures at Elevated Powers

A partial analysis has been made of the possibility of nucleate boiling in the fuel of the cycle VI core at higher power.

The results of this study indicate that the ORR fuel elements may be run at 45 Mw with inlet temperatures of 120°F and water flows of 18,000 gpm without danger of nucleate boiling. It further indicates that shim rods (especially new ones) will probably cause nucleate boiling under the same conditions. At a flow of 20,000 gpm the case of the shim rods becomes marginal. No results are yet available on the effect of increased pressure and thermal stresses under these conditions.

Decontamination Scrubber

The ORR decontamination scrubber system has been modified to accomplish the following:

1. A "fail-safe" control system; i.e., a control system that will not change aspect as a result of power failure.

2. A "test" made to permit a complete operational test of the scrubber without shutting down the building ventilation system or operating the steam fan.

3. A more reliable electrometer that will not energize the scrubber as a result of false signals. This is set to actuate at ~80 mr/hr. No basis exists for this figure except the fact that it would be necessary to evacuate if air activity reached this level in any case. The effect on ground activity is not known since no levels for stack release are available.

4. Continuous ventilation for the basement cells.

The "fail-safe" control system utilizes a unique combination of relays and contacts and is dependent upon the emergency generator's providing emergency power in less than 10 seconds after a failure of normal power.

The "test" mode permits a functional check of the scrubber without shutting down the building ventilation system or starting the steam fan. Even in the "test" mode an alarm condition by the electrometer will automatically take the scrubber out of "test" and initiate scrubber operation for emergency conditions.

The alarm-seal in the electrometer has been removed from the circuit, thereby eliminating the use of the electrometer reset button. A high radiation condition must persist for at least 10 seconds to energize the scrubber through the electrometer. Once the scrubber is energized it will continue to operate until it is reset by pressing the "reset" button. False signals resulting from voltage fluctuations will not energize the scrubber because a lock-out feature in the electrometer discriminates between real and false signals. The lock-out feature automatically resets itself. An alarm is received through the scrubber annunciator under the following conditions: (1) power failure to the electrometer, (2) during a lock-out condition, and (3) during a high-level radiation condition.

In addition to the duct work that traverses the building to provide emergency ventilation, several new ducts have been installed connecting basement cells to the scrubber duct. The damper between the scrubber and the stack has been permanently opened, thus, providing continuous cell and building ventilation through the scrubber. Since cell and emergency ventilation must be assured at all times when the reactor is operating, a flow switch will be installed in the duct. The switch will be connected to the reactor controls and will initiate a reactor setback if the air flow in the duct drops to a predetermined value.

Since more than 5000 cfm of air is continuously exhausted from the building through the scrubber, the building could be placed under excessive negative pressure should all the roof exhaust fans be turned on simultaneously. Therefore, three of six exhaust fans have been locked out and must not be used while air from the building is being exhausted through the scrubber.

A complete operational check of the scrubber is made every four weeks while the reactor is shut down.

Emergency Generator

The emergency diesel-powered generator has performed satisfactorily during the past year except for one occasion when the automatic transfer switch failed to transfer the load to the emergency system. This failure was attributed to malfunctioning of the Woodward governor due to the lack of lubrication. The lubrication of this unit has been placed on a weekly check list. There still is not sufficient load on the emergency power circuit to test operate the diesel routinely without providing additional load. The additional load is obtained by placing the entire ORR building power demand on the diesel for a period of approximately five hours during each regularly scheduled shutdown. It has been necessary to do this at night when welding machines were not being operated.

Pipe Chase

Rupture of experiment or reactor water system lines in the pipe chase under the reactor would be a serious event. A large number of lines are located in this area in very close quarters. A violent rupture or explosion in one line may damage several others. The lines in the pipe chase are installed so close to each other that inspections, maintenance, and new installations are difficult to perform. Sensors for some vital reactor process instrumentation are located in the pipe chase. They could be accidentally damaged by personnel working in this area or by some other unexpected occurrence. Experiment lines exposed in the pipe chase are not shielded. These lines may become highly radioactive, seriously limiting working time in the pipe chase. The cost of providing three inches of lead shielding for 2-inch lines has been estimated at $65 per foot and experimenters have been asked to shield their lines where necessary.

Core Changes

During the past year several core configurations have been used.

Some recent requests for core changes which will require a great deal of study are:

1. The desire of many beam-hole experimenters to locate a fuel element directly in front of their facility.

2. A request to increase the flux by a factor of 4 in the north and south facilities.

Fuel Storage

As reported in the last review, fuel is stored in the ORR pool in critically safe storage racks providing a maximum capacity of 120 elements. All fuel movements are by orders of the reactor supervisor. In addition to the regular storage rack, a gamma-grid irradiator is being placed in use. This unit will use the "hot" elements immediately after they are removed from the reactor. In the grid the elements are placed in individual holders which contain cadmium curtains. A. D. Callihan has reviewed the intended configurations and approved them.

Fuel Burnup

To date the average percent burnup is about 31%. This low figure is due, in part, to the low weight elements which were on hand for the initial nuclear tests.

Hot Cell

Usage of the hot cell is increasing as more experiments are being removed from the reactor. It is proving very useful in making preliminary inspection of experiments, and it has also been used to obtain temperature data on irradiated ORR fuel in air.

Modifications which make the cell more effective are the addition of lead doors on the west side, the installation of lead brick shielding around the access ports, and the procurement of several tools.

JACox

J. A. Cox

JAC:gc

Attachments

cc: A. F. Rupp
A. M. Weinberg

APPENDIX A

Status of ORR Power Level Increase
F. T. Binford

The proposal to increase the power of the ORR has associated with it eight basic problems. These problems, all of which are interrelated, are listed below.

1. Adequate heat removal must be provided in order to prevent the occurrence of excessive temperature in the reactor core.

2. External heat exchangers of sufficient capacity to remove heat at the increased power level must be provided. This includes some provision for maintaining the temperature of the water circulating through the demineralizer low enough to prevent damage to the resin. It is also necessary to make sure that any increased heat load in the pool is provided for.

3. The effect of increased power level on the heat generation in the reactor structure due to photon absorption must be examined. This is necessary to determine whether any significant thermal stresses will arise due to increased temperature gradients.

4. The upper limit of the allowable pressure stresses in the reactor vessel and piping must be determined in order to set the upper limit of coolant flow.

5. The efficacy of the present biological shielding to provide adequate protection at elevated powers must be assessed. In particular, the distribution of activity in the cooling system and in the pool under altered conditions of flux, temperature, and flow must be considered.

6. The effect of the increased power level on experiments must be investigated.

7. The applicable fuel cycle for the new power level must be investigated; and, if heavier loadings are indicated, the need for additional control elements should be assessed.

8. The necessity for a more positive emergency shutdown system.

The present plan is to bring the reactor to 30 Mw at as early a date as possible. To accomplish this it is necessary to consider only items 2, 6, 7, and 8 above. Because the reactor was designed for 30 Mw and because more than a year of experience at 20 Mw has demonstrated that it operates essentially as designed, there is no reason to believe that it will do otherwise at 30 Mw.

The external cooling is, and always has been, inadequate for 30 Mw operation; and the purchase and installation of sufficient heat-exchanger capacity to permit operation at this level has already been initiated. Since it is not contamplated to change the flow and heat flux conditions within the core from those proposed in the original design, it seems logical to conclude that no significant change in temperature or pressure stresses will be experienced, nor will any modification of the internal heat-transfer requirements be necessary. The biological shielding was designed to handle 30 Mw operation; although, as originally intended, it will be necessary to make sure that the upward transport of N^{16} in the pool is sufficiently inhibited.

The effect of the increased power level on experiments is already under investigation, and it is my understanding that each experiment will be reviewed to determine what, if any, modifications are necessary to prepare them for operation at this power. At 30 Mw operation the rate of fuel burnup will exceed the present rate by 50%. This means either that the core will have to contain more excess reactivity at the start of each cycle, or that some modification will have to be made in the present reloading schedule. Reflector-poison rods which would permit the loading of additional excess are presently under design. Until these are ready for use it will be necessary to shut down for reloading earlier in the cycle (probably after 1 1/2 weeks) than is now done. The exact loading pattern has not yet been worked out.

There has been considerable debate as to both the necessity for, and best method of, providing emergency cooling to remove afterheat following shutdown. The original approach-to-power tests indicate that the reactor can be sufficiently cooled by natural convection alone up to at least 17.5 Mw. It seems likely that this range can be extended; however, it has been decided that some form of emergency shutdown cooling, more positive than the present standby pump, should be provided. Two proposals, one for a constantly operated pump driven by an internal combustion engine and the other for a battery operated standby pump, have been studied by the Engineering Department. To date no decision as to which system will be used has been made.

The long range objective is to increase the power level of the ORR to the maximum value practicable without major revisions in the reactor system. The factors which appear most likely to control the magnitude of the upper limit to which the power can be raised are the thermal and pressure stresses which can be tolerated in the reactor structure and in the reactor cooling system. The maximum permissible values of the pressure stresses establish the upper limit of the coolant velocity and hence determine the maximum heat-transfer coefficient which will be available to remove energy from the core components. The maximum permissible values of the thermal stresses will dictate the upper limit of coolant temperatures which can be utilized. Once these values have been obtained it becomes possible to determine the design characteristics of an external cooling system optimized to remove the maximum heat at the lowest cost. The problem of obtaining realistic values of these stresses has been assigned to the Reactor Projects Division.

The original design specifications of the ORR contemplated a maximum fuel element surface temperature of 210°F. Since this value is approximately 45°F below the saturation temperature of the water at the lowest pressure encountered in the core, it is now believed that it is perfectly reasonable to operate the reactor in such a manner that the maximum surface temperature of the fuel does not exceed this saturation pressure. As a first approximation this would represent an increase in permissible heat flux of 1.6 under conditions of flow identical with those originally contemplated. This advantage is partially offset by the existance of some evidence that the maximum-to-average heat flux ratio exceeds that contemplated in the original design. In a series of calculations based upon neutron flux measurements and assuming uniform burnout, T. E. Cole has found that, while in general the heat flux peaks occurring at 20 Mw are less than those originally expected, the heat flux at some locations in the control rods exceeds by as much as 40% the predicted value. It is believed that by appropriate redesign of those control elements such heat flux peaks can be eliminated, and thus the full advantage of the increased plate temperature can be relayed. Studies aimed at obtaining a better understanding of this problem are at present being made by members of the Operations Division.

Correlation of the results of the foregoing work should make it possible to establish criteria for operation at the highest level consistant with our present thinking with regard to heat removal if adequate external heat exchangers are provided.

As in the case of 30 Mw operation, attention must be given to the effect of increased power on the various experiments. While definite conclusions cannot be reached until the upper limit of power is established, it would appear prudent to anticipate the power increase by taking this fact into consideration in the design of new experiments. A power level of 45 Mw suggests itself as a reasonable design point.

If, in making the change from the present to higher power level, it happens that the temperature of the coolant stream sent to the demineralizer is increased, it will be necessary to examine the effect of such a temperature increase on the demineralizer resins. The present material is known to deteriorate at elevated temperatures. The solution to this problem is quite simple and requires either the installation of an auxiliary heat exchanger or the use of resins which are not damaged at the temperatures involved.

Any change in power or in coolant temperature will be reflected by a change in heat removal requirements in the pool system. A series of experiments, utilizing variations in coolant temperatures and power, has been planned. It is expected that the effect of such changes on the pool heat load can be determined by this means.

As pointed out in the discussion of the 30 Mw case, any change in power level will result in a change in the fuel cycle, and this will have to be provided for as stated earlier.

The present biological shielding was designed quite conservatively for use in connection with 30 Mw operation. It is believed that, with few exceptions, it is adequate for power levels considerably in excess of 30 Mw. However, before any major program to increase the power substantially above 30 Mw is contemplated, the adequacy of the shielding should be reviewed in the light of the past operating experience. The N^{16} suppression problem will certainly require attention, and there will probably be some points at which additional local shielding will be required. Although all of the experience so far indicates that no problem exists, it will be necessary to re-evaluate the gamma heating in the concrete shielding to make sure that no significant stresses will develop.

The shutdown cooling problem closely parallels that which arises in connection with 30 Mw operation.

Finally, one can consider the hybrid problem of obtaining power in excess of the present level solely by permitting an increase in the coolant temperature. The areas in which investigation is required include those concerned with thermal stresses, heat removal in the pool, demineralizer temperatures, and N^{16} transport in the pool. The upper limit on plate temperature can be considered, as before, the saturation temperature at the lowest core pressure. The Reactor Projects Division is investigating the stress problem and the Operations Division is considering the others.

(COPY)

APPENDIX B

To: F. T. Binford Date: November 10, 1959

From: M. W. Rosenthal

A statement of what we are doing and plan to do with regard to the ORR is enclosed for use in your operations review.

Sgd. M. W. Rosenthal

M. W. Rosenthal

MWR:sh

cc: R. A. Charpie
J. A. Cox
A. P. Fraas
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File

Personnel of the Reactor Projects Division are investigating several problems relating to two changes which may be made in operating conditions of the ORR. These are associated with: (1) a proposal to increase the present power removal capability by increasing the temperature level of the reactor coolant, and (2) the plan to increase the power capability by the installation of additional heat-exchange surface.

1. Increasing the water temperature would permit a higher rate of heat removal from the present air coolers. If found to be permissible, this method of operation would probably be used to allow continued operating at the 20-Mw level, even in hot weather, until additional heat-exchange equipment is provided. The major question regarding the proposal is whether expansion stresses in the reactor vessel and the piping system become excessive at the water temperatures desired. Piping stresses are being computed at those sections where they may be too high. The questionable areas have been identified as in the exit water lines, the line between the pumping station and the air coolers, and the pool interconnection lines. The results of the pipe-stress analyses should be available within several weeks.

A more difficult problem is estimation of stresses in the obround large-facility tubes which result from vertical expansion of the reactor tank. Analysis of the stresses in these members has begun, but because of the complexity of the problem the stress calculations will not be very reliable. Unless a conservative evaluation of the analysis indicates the system to be safe, further investigation involving strain measurements on the reactor and on a model is planned.

2. Assuming that a sufficiency of heat-exchange equipment is supplied, the condition which restricts the power of the ORR may be excessive surface temperatures in the core. In this case, the allowable power level could be increased by increasing the flow rate of water through the core. However, the greater flow rate would result in higher hydrostatic pressures in the reactor tank than now exist. A significant flow increase would be accompanied by pressures which are higher than presently allowed for the core region and would possibly result in excessive stresses in the flat plate adjacent to the pool-side facility.

This condition will be investigated by analysis of the stresses which occur as a result of increased power. As with the obround sections, strain measurements on the reactor tank and on a model will probably be required. The situation with regard to thermal stresses which could result from radiation heating of the reactor vessel will also be examined.

X-847

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

December 1, 1960

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Subject: Information for Review of Oak Ridge Research Reactor Operation

From: J. A. Cox, Prepared from Reports by W. R. Casto, F. T. Binford, R. A. Costner, Jr., S. J. Ditto, R. V. McCord, W. H. Tabor, and J. F. Wett

During the past year the power of the ORR was increased to 30 Mw, and the new cooling system installed last summer will permit operation at 30 Mw the year round. Work is under way on increasing the reactor power to about 40 Mw, and a proposal for this will be completed fairly soon. Aside from increasing the reactor power, the major difficulty during the past year has been in reviewing the hazards of experiments and in obtaining sufficient supervisors to replace those who terminated or transferred to other positions. A further difficulty has been encountered in emergency power equipment. Four different emergency power devices are now installed, primarily to supply shutdown cooling flow for the reactor core. A significant number of failures has occurred in all devices. This does not include the battery-Mg set installed for experiments.

The measures adopted for controlling corrosion of aluminum pipe embedded in concrete and the aluminum pool liner appear to have been effective, and the corrosion rate of samples has been satisfactorily low.

An operating manual (ORNL CF-60-8-46) has been completed, and this is a great aid to training and to operatinn in general.

Manpower

It has been very difficult recently to obtain replacements for shift engineers who have transferred to other jobs, and as a result the ORR is now being operated with several day supervisors sharing one of the shifts. Also it has been very difficult to obtain replacements in time to give sufficient training before they take over an independent job.

The constant increase in numbers of reactor operation trainees has also made the engineers' jobs more demanding. The present schedule of trainees includes 5 foremen and 8 engineers from TVA, 6 engineers from South Africa, one engineer from Spain, and next August or September a new group from the ORSORT training course will arrive. It is believed that in order to give better continuity for the trainees and for our own people that one trained supervisor is needed who will direct the training on shifts, assign problems to different trainees, and hold classes at intervals where all the trainees are brought together from the different shifts thus strengthening the program. There is a danger, with the present system, that a wide variation will exist from one shift to another, depending on the agressiveness and training of the shift engineer.

Experiments

Pool Experiments

The installation of in-pool lines for experiments has reached the point where th space is extremely limited as indicated by the attached photograph.

Beam-Hole Experiments

There are presently four beam-hole experiments, and the external equipment for the remaining two beam holes is nearing completion. One major change in the operating experiments required the replacement of a beam-hole plug collimator due to a plugged inlet water line. It was found that creeping of a gasket had caused the stoppage.

In-Pile Experiments

Six major experiments utilizing the access flanges through the tank top have operated successfully. Two new experiments are being reviewed by the Operations Division, and another is undergoing modification to comply with containment and maximum pressure criteria.

Engineering-Test Facilities

The GCR Loop No. 2 experiment and associated equipment occupying the south engineering-test facility is in the early stages of construction and is scheduled for operation in the third quarter of 1961. The 5-ft diameter outer plug was removed and replaced with the outer plug for GCR Loop No. 2. The CPFF contractor is currently constructing the containment cell and making necessary modifications to the building and building grounds.

Reactor Controls and Shutdowns

Instrument Maintenance

Probably the most important change that has contributed to fewer shutdowns from instruments is the new checkout procedure used after a long shutdown. This procedure, worked out by S. J. Ditto of Reactor Controls, affords a "live" or "hot" check of each recorder switch with its associated relays and circuits. A malfunction or improper sequence of events is readily detected and corrected. This has been part of a general maximum effort to keep the recorders in excellent mechanical condition. They also are checked at each midcycle shutdown.

More emphasis has been placed on checking of tubes and components in the safety equipment. No circuit changes or design alterations are made without issuance of approved ORR change memoranda and a follow-up check of these changes by Reactor Controls.

Miscellaneous Changes

The position of the shim rod controlled by the servo system was changed from position B-4 to D-6 which is a higher flux position. Because the shim rod in the new position is worth more per unit length, the servo can control the reactor power more easily. Installation of the improved, travelling, limit switches was also completed, resulting in a change of their location from the subpile room to the control room. The Bodine motor driving these limit switches was replaced with a new motor 33% slower. This permits more uniform rod speed withdrawal during startup.

Since the beginning of ORR operation, it has been an administrative procedure to operate the reactor with no less than two level safeties in use. In order to augment this procedure with an electronic control, the following changes were made in the safety system. Upon failure (or indicated failure) of one channel, a "safety trouble" annunciator is energized and cannot be cleared as long as the trouble persists. In order to annunciate the failure of a second channel, a "two safety troubles" annunciator is provided which is energized upon failure (or indicated failure) of two safety channels. A shim rod reverse is initiated simultaneously.

For additional operational continuity and safety, three further changes were made. One provided an individual key for each experiment, tie-in, "E"-panel switch. Another provided two reactor water exit temperature recorders. Monitoring of outlet temperature on continuous recorders instead of a multipoint recorder as previously used improves the safety of the system. The third change provided an additional Log N channel. This channel is identical to, and independent of, the original channel up to the output of the period amplifier. A multistage, SB-1 switch can be used to select one or the other channel, but not both simultaneously, as the "operating" channel. Only the channel thus selected is connected to the reactor control and safety system and the safety trouble monitor circuits associated with the period sigma amplifier. In addition, the contacts on the SB-1 switch which are used for the control and safety system transfer are of the "make-before-break" type; and, although only the selected channel has any control function, the signals to the recorders of both channels are unaffected by the switch.

30-Mw Instrumentation

To accommodate the additional instruments needed for 30-Mw operation, the graphic instrument panel was extended, requiring major relocations of existing instrumentation.

Nuclear instrumentation supporting the 30-Mw operation includes: (1) dual Log N channels; (2) dual ΔT channels on reactor water; (3) dual reactor water exit temperature monitors; and (4) necessary components to include the two Be-Cd shim rods in the fast scram circuit.

The necessary adjustments in ionization chamber position and trip points on process instruments which are connected to the safety circuit were completed without incident.

Shutdowns

Malfunctioning of reactor controls has contributed to reactor shutdown time. Failures causing reactor shutdown or prolonging reactor startup have occurred in rod-drive components and ionization chambers. As a result of the failures in the chamber circuits, a routine replacement of each unit on a yearly basis has been established.

An associated component of the control rod drive, the upper hold-down are which houses the upper shim-rod bearings, was the cause of an ~two-day shutdown. Comprehensive tests were performed which diagnosed the trouble to be in this unit. Inspection of the unit indicated excessive clearance in the bearings. A new unit was installed. A review of the whole shim-rod system including the hold-down arms is under way in an attempt to make these more reliable and less susceptible to failure.

TABLE 1. ANALYSIS OF UNSCHEDULED SHUTDOWNS

October 1959 - September 1960

Description	Number	Down Time (hr)
Real Cause	1	0.233
Instrument Failure		
Operations	5	12.167
Experiments	15	3.465
Subtotal	20	15.632
Power Failure	4	2.017
Mechanical Failure, Reactor Controls	10	54.702
Human Error		
Operations	3	0.450
Experiments	1	0.017
Engineering and Mechanical	2	0.616
Instrumentation and Controls	1	0.183
Subtotal	7	1.266
Flooded Instrument Sump	1	9,583
TOTAL	43	83.433

TABLE 2. CAUSES OF SHUTDOWNS DUE TO EXPERIMENTS IN ORR

October 1959 - September 1960

Date	Experiment	Cause
1-9-60	MSR	Setback due to failure of sample outlet temperature recorder. While starting the repair procedure, the recorder was manually run upscale instead of down. Failure was corrected by adjustment of sensitivity.
2-14-60	MSR	Setback due to failure of electronic tube in instrument monitoring temperature of in-pile pressure pipe.
3-7-60	MSR	Setback due to electronic tube failure in the sample inlet temperature recorder.
3-13-60	EGCR (poolside)	Setback due to electronic tube failure in the recorder monitoring temperature of capsule No. 4.
4-9-60	GE (F-2)	Reverse from failure of thermocouple No. 8 on TR-6. Thermocouple was replaced with a spare. The "backup" reverse contact was apparently set slightly downscale of setback contact resulting in premature reverse.
4-25-60	GE (F-2)	Real Cause. Setback (and "backup" reverse) due to failure of air regulator in the main supply line, resulting in complete loss of cooling flow and in high sample temperature. The exact reason for the failure has not been established; however, a considerable amount of foreign matter was found in the control air supply regulator.
4-26-60	GE (F-2)	Setback due to failure of thermocouple monitoring specimen temperature. The thermocouple was replaced with a spare.
5-3-60	Solid State (B-9)	Setback due to failure of thermocouple monitoring specimen temperature. The experiment was withdrawn from the flux and thermocouple replaced with a spare, but Operations requested a second change since experimenter used thermocouple with no operating history for first change.

Table 2. Continued

Date	Experiment	Cause
6-8-60	MSR	Setback due to failure of electronic tube in inlet temperature recorder.
6-22-60	EGCR (poolside)	Scrams (2) resulted when power was cut off to capsule No. 1 instrumentation. No experiment was being run and E-panel switch was in the "test" position. After the second scram, a broken lead on the S-2 contact of the switch was discovered and corrected.
6-29-60	GE (F-2)	Setback resulted from human error when, in attempting to replace a thermocouple on an "alarm only" circuit, the wrong thermocouple plug was pulled and then replaced.
7-5-60	GE (B-1)	Setback resulted from upscale spike of recorder. About 24 hrs later a spike of much shorter duration caused another setback which did not result in a shutdown. Cause was undetermined trouble in the recorder, but is believed to have been caused by electronic tube which failed under load at long intervals.
7-6-60	GE (F-2)	Setback caused by failure of thermocouple.
7-7-60	GE (B-1)	Setback and "backup" reverse resulted from human error while changing out erratic recorder (see 7-5-60). A jumper was placed across two terminals for the change and two wires had the same identifying numbers. The wrong wire was selected.
9-16-60	GE (F-2)	Setback and "backup" reverse caused by failure of thermocouple which was replaced with a spare.
9-17-60	GE (F-2)	Setback and "backup" reverse caused by failure of thermocouple which was replaced with a spare.
9-22-60	EGCR (poolside)	Setback caused by failure of thermocouple (capsule No. 6) which was replaced with a spare.

TABLE 3. RELEASE AND DROP TIMES OF ORR CONTROL RODS
(All Times in Milliseconds)

	Rod No.	5-20-60	6-16-60	7-28-60	8-13-60	9-14-60	9-30-60	11-30-60
Release time	1*			19.5		21	15.3	
Time of flight	1*			270		270	280	
Release time	2*			12.0	18	21.8	22.6	
Time of flight	2*			250	260	260	300	
Release time	3	17	18	12.0	21	21.2	17	15.5
Time of flight	3	310	270	260	275	270	250	160
Release time	4	19.5	21	9.7	15	18	16	17.7
Time of flight	4	300	290	270	280	290	280	260
Release time	5	18.5	20.5	7.0	17	13.8	12.7	12.0
Time of flight	5	310	290	260	305	270	270	120
Release time	6	21	18	8.8	17	14	27	19.6
Time of flight	6	300	290	240	265	280	220	260
Water flow, gpm		~16,000	~16,000	18,700	~18,500	~18,500	19,600	18,800
Rod Position, in.**		3/20	3/15	3/20	3/20	3/20	3/20	3/20

*No reactivity credit assumed in loading; therefore, checks have not been made each month.

**Position for release-time check/position for time-of-flight check.

Reliability of Equipment

If the shutdowns due to experiments are examined, it will be seen that a number of these were due to failures of recorders. A system of safety instrumentation for experiments and for the recorders has been proposed whereby the recorders are left out of the safety circuit and the safety action is provided by a reliable amplifier. This would have several advantages.

1. The instruments would be much smaller.

2. A separate panel of moderate size could contain all of the safety instrumentation for experiments and perhaps even for a reactor.

3. The safety instrumentation could be separated from other instruments required for data collection or other purposes.

4. Amplifier units could be the plug-in type so that they could be replaced immediately rather than having to trouble shoot in large, complicated instruments which could not be removed from the system.

5. Such a system would lend itself to a two-out-of-three coincidence system through reduction of size and simplification.

6. This type of system would probably be much cheaper to build.

7. The present designs make no attempt to separate safety instrumentation from other types of instruments, and a great deal of confusion exists as to the reliability required in each section of the system. By making a clear-cut separation, the reliability can be emphasized where it is needed.

Exchange of Information with Harwell

Arrangements have been completed to exchange information on operating reliability of instruments, reactor components, and emergency equipment with Harwell. The first such exchange should be ready soon. We propose to describe the troubles which have occurred both with the reactor and with experiments which have resulted in shutdowns and to request similar information from the British. It is understood that the British have adopted a system similar to that mentioned above in which recorders are eliminated from safety circuits along with a considerable amount of two-out-of-three circuits to prevent spurious shutdowns. Comparison of the ORR and the British experience should be quite interesting. Also our failures with emergency power equipment will be described, and the British will be asked to describe their experience.

Cd-Be Control Rods

The Cd-Be control rods installed last summer in lattice positions F-4 and F-6 have not operated properly. They have sometimes failed to seat properly as if a bearing were binding the rod. The rods fall freely until they get 2 - 3 inches from their seat, but because of this trouble we have not taken credit for them in loading the reactor. A series of tests and various test devices are being designed to disclose the cause of this trouble in addition to tests described in Table 4.

TABLE 4. TESTS* OF BE-CD RODS
AND ASSOCIATED LATTICE POSITIONS

Test	Object	Results
To determine if Be-Cd rods function properly in fuel-Cd rod positions.	Determine of Be-Cd rods are straight.	Proper function observed.
To determine if fuel-Cd rods function properly in Be-Cd rod positions.	Determine if Be-Cd rods are straight.	No improvement indicated from difficulty observed with Be-Cd rods.
To determine if special rod with "floating piston" shock-absorber section functions properly in Be-Cd positions.	Determine if possible misalignment of drive tube and bearings.	Some improvement indicated from difficulty observed with Be-Cd rods.

*At the time of the above tests one of the lower plug shock absorbers was an original unit and one had been installed only a month previously.

The two new Be-Cd rods, previously described, were calibrated simultaneously against the four fuel-Cd rods. The total worth of the two rods is 2.9% $\Delta k/k$. To date, no reactivity credit has been taken for these two rods. Although no difficulty has been experienced under full-flow conditions, drop tests with no flow showed that both rods cannot be relied upon to fully seat. A number of attempts have been made to identify the cause of the difficulty. The results are shown in Table 4. As can be seen from this information, no positive problem area could be identified; although there is some indication of possible misalignment of bearings. This problem has been discussed with E & M personnel who are investigating methods of checking alignment of critical points.

Investigations have been made on the feasibility of using Al-Cd in place of the fuel-Cd rods. In one test the Al-Cd rod was compared to three fuel-Cd rods (containing 131, 78, and 35 gm of U^{235}) in the same core position (D-6). During each test the reactor was brought to critical with the other rods in their normal locations at the same position. This test showed the differential reactivity of the Al-Cd rod to be slightly less than that of the 35-gm rod. Also the Al-Cd rod had to be withdrawn about an inch farther than the 35-gm rod to maintain criticality. At the same time, flux measurements were made to determine the peaking results from "flux trapping" in the water gap between the fuel section and the cadmium section of the fuel-Cd rods. This is most important in its effect on an adjacent fuel element. It was found that for shim rods "burned up" to below 80 gm of fuel, the resultant flux peak may limit future increase in reactor power.

At the present time, it is planned to investigate the possible increase in core life resulting from increased frequency in shim-rod replacement. while a greater usage of control-shim rods will increase operating costs. This will certainly insure high fuel element burn-up and probably reduce the necessary fuel element inventory. It will also increase the length of operating cycles since heavier shim rods will permit a heavier fuel loading.

Radiation Incidents

There have been five "unusual occurrences" at the ORR during the period covered by this report. Four of these resulted in appreciable increases of the normal background.

1. On 6-27-60 two research men entered the B-9 cubicle without informing Health Physics or Operations personnel. Both men wore contamination-zone clothing but no gas masks; one carried a cutie-pie survey meter. Before entering the cubicle they checked the CAM monitoring cubicle air and, once inside, the background radioactivity monitor. Both instruments were reading normally. During the equipment check which involved valve manipulations, they noticed that the cutie-pie was increasing from 20 mr/hr to 60 mr/hr and that the inside background monitor was increasing also. They immediately left the cell and found that the CAM was off-scale but had failed to alarm. This instrument was reported to have been working properly during a calibration check three days prior. The filter on the CAM which was removed probed approximately 1 r/hr at 2 inches. A nasal swab from one man counted approximately 800 d/m (β-γ) above background; one man had a spot on his hand which probed 2 mr/hr.

Analysis showed Cs^{138} to be the predominant emitter, and the experiment was put under off-gas to reduce the radiation hazard inside the cell and to reduce the probability of a further release. Surfaces outside the cubicle were checked to determine that no contamination had been spread.

As a result of this incident the cubicle entry rules now require Operations and Health Physics notification and approval, and face masks are required. In addition, this incident points to the fact that since radiation monitoring instruments are not installed in duplicate, as are most reactor and experiment instrumentation, they must have much more frequent calibration and checkout. Except for the fact that the personnel involved interpreted the increase in background as a possible increase in air radioactivity, they could have inhaled a dangerous quantity of radioactivity.

2. On 6-11-60 during removal of a sample from the General Electric air-cooled fuel-testing loop, a considerable portion of the third floor and several other areas of the ORR Building were contaminated.

The sample is contained in a flexible metal hose which extends from the top of the permanent tube to the core region. In addition to being an air pipe, this hose also serves as a conduit for heater and thermocouple leads. The interior of the hose becomes very contaminated due to radioactive dust and fission products carried from the core region by the exhaust air. Normal removal procedure involves, in part, unscrewing a coupling between a "Y" section of tubing and the flexible hose. On this occasion, the coupling could not be disconnected so the hose and wires had to be severed below the coupling with hand-operated cable cutters. The hose section was pulled into a carrier, but about 12 feet of hose protruded above the carrier and another cut was necessary. Neither cut could be made in an enclosure connected to off-gas; and, as a result, particles dislodged from the hose were dispersed over the general working area. Also the contaminated cutters were not carefully handled and contributed to the contamination spread. The areas affected were the working bridge, third level pool balcony, west truck entrance, and isolated spots on the first and third floors.

The factors which contriubted to the incident were primarily procedural, and detailed procedures for experiment removal have been improved.

3. A small fire occurred on 8-18-60 in the south hot cell at the ORR during disassembly of GCR-ORR capsules. The fire apparently involved alcohol which had been added to the mixture of mineral oil and alcohol provided for covering drops of NaK. The fire was quickly extinguished with the Ausul extinguisher. No damage was done to the cell nor was any radioactivity released. A CAM which was sampling the cell air indicated no increase in activity. Experience gained from this incident is helpful in making plans for handling future experiments.

4. Evacuation of the ORR Building was caused by radioactive gas which leaked from a GCR-LITR capsule on 8-23-60. The sample, after removal from the LITR, had been brought in a carrier to the ORR for subsequent transfer to the south hot cell. The CAM's indicated a general increase in air activity in the ORR Building, and at about 10:45 a.m. the building was evacuated. After the carrier was recognized as the source of activity, it was wrapped in plastic film and connected by a hose to the off-gas system. Air activity declined, and operations in the building were resumed about 11:46 a.m. The sample was transferred to the hot cell on the 4-12 shift without incident.

As a result of this incident, entry into the building of all loaded carriers requires the approval of the ORR shift engineer and written approval of the Operations Division Radiation Safety Officer for hot cell work on the material contained in the carrier. Readily accessible off-gas connections have been provided at the west truck entrance as a further precaution.

5. Contamination of the pneumatic-tube hot laboratory and the stairway leading from the laboratory to the second floor occurred on 10-31-60. Investigation by Operations and Analytical Chemistry Division personnel revealed that a capsule containing sodium carbonate was probably not sealed properly prior to irradiation, continuous radiation monitoring instruments were not provided, and the individual involved was an ORINS student not experienced with routine methods in handling samples. As a result of this incident, experienced personnel will supervise such operations in the future, and continuous monitoring instruments have been requested but are not yet on hand. During the interim, monitoring by hot-laboratory personnel will necessarily be relied upon.

Recent Troubles

An unusual number of difficulties were encountered with the ORR control rods and related components during the recent shutdown--November 20 through November 30, 1960. Startup of the reactor was delayed approximately four days while repairs were being made. A brief description of these troubles follows.

11-25-60

During the 4-12 shift the drive motor of the No. 6 shim-rod drive unit burned out when the drive unit was unintentionally driven through the upper limit causing the motor to stall. Investigation revealed that an error in wiring during replacement of a relay was the cause of the malfunction. Contact points on the relay were connected in the normally closed condition instead of the normally open condition actuating the motor circuit.

11-26-60

The shim rod in D-4 position was damaged during its removal from a stuck condition in the hold-down arm. The rod had become stuck when the hold-down arm was lowered over the unseated rod. After other methods failed the rod was freed by engaging the cross of the drive unit and turning the motor by hand to pull the rod down. The hold-down arm was removed from the reactor vessel and inspected under about 6 feet of water. Visual inspection indicated no apparent damage.

11-27-60

During the instrument check the shim rod in D-4 position would not drop when the magnet current was reduced to zero. Inspection of the push rod, with the magnet removed, revealed a loose component (a nut). The unit was reassembled. Checks were again performed, and the rod failed to drop with zero magnet current. This drive tube was replaced on 11-28-60.

11-28-60

Difficulty was experienced in obtaining normal operation of the shim rod in core position D-6. The rod would not drop with any reliability when the magnet current was reduced to zero. The drive tube was replaced.

11-29-60

A short developed in the magnet leads which supply the shim rod in core position B-4. With the magnet removed, the push rod unit was checked and found to be sticky. The drive unit and magnet were replaced.

11-29-60

Instrument checkout revealed a fail-to-drop on the shim rod located in core position B-6 when the magnet current was reduced to zero. Inspection of the push rod indicated intermittent sticking. The drive unit was replaced with one of the units previously removed after it had been repaired.

Inspection of two of the faulty drive units indicated excessive wear and indentations on the plunger and the presence of foreign matter similar to lubricant on the body of one of the plungers.

From the observations made to date it appears that several improvements can be made in the design of the ball mechanism of the drive units. First, by cutting spiral grooves in the body of the plunger (which operates the balls) grit and dirt may clear itself instead of causing the plunger to stick. Second, the slots in the wall of the drive assembly are of such a shape that the balls appear to stick in these slots. This may be the reason for the wear or indentations on the plunger seat where the ball contact is made. By tapering the slots upward it may be possible to eliminate the resistance to movement of the balls.

Finally, the recent troubles indicate the potential hazard of a jammed shim rod. Aside from other hazards a jammed rod may coincide with a failure of controls which causes a shim rod to drive upward. This could exert tremendous force on the core, even damaging the bearing and grid assemblies. An effort will be made to design a clutch or other torque-limiting device for the drive units to lessen this hazard. A similar hazard exists in the LITR-MTR design, but there the magnet assemblies fail before damage occurs to the core. Unfortunately, this does not appear to be the case in the ORR.

Jamming of a shim rod can occur from failure of the hold-down arm lock as occurred last summer. Damage to the core could easily occur in such an event. since the hold-down arm hinges upward binding (and bending) the shim rod. A design study is under way to make the locking device more reliable, but it is felt that a torque-limiting device is also extremely important for protection of the core.

Fuel Loading Calculations

A program is being developed which will permit the fuel loading to be calculated on the ORACLE or other computer, and the advantages of a fuel element containing more fuel than the present 200 grams is being studied.

New Fuel

An order has been placed for three hundred 200-gram fuel elements at a cost of approximately $554 apiece. As soon as time permits specifications will be written on shim rods so that these can be procured from private industry. This may take considerable time with the present staff available.

Fuel Handling, Storage, and Shipment

One of the problems at the ORR is the possibility of damaging the aluminum pool liner from heavy carriers hanging in the pool. Since the pools are so deep, it is not considered feasible to set the carriers on the pool floor. Instead they must be suspended from the 20-ton crane during operation. While the present fuel shipping container weighs only 7 tons, we have avoided putting this in the pool and, instead, individual fuel elements have been shipped to the Graphite Reactor canal in a single-place container. Since adequate transportation is not available and since the small container must be used seven times as often as the seven-place container, it is now planned to load the larger container in the ORR pool.

Emergency Equipment

Emergency equipment at the ORR has been disappointing in that a number of failures have occurred. Fortunately, none of these failures have occurred at a time when emergency power was badly required. Failures occurred on at least four different occasions with the diesel generator, and in some of these cases several malfunctions existed at the same time so that it was difficult to determine whether one or all were responsible. The gasoline-driven pump has failed to start rather frequently and has actually stopped while operating on at least two occasions. The battery-powered D.C. motors on the main water pumps have also failed several times--apparently due to undersized chargers which have failed and allowed the batteries to become discharged.

Diesel Generator

On 8-19-60 the diesel engine failed to start as a result of a discharged battery. An investigation revealed a blown fuse in the battery charger in addition to a shorted cell in the battery. Replacement of the battery and the blown fuse remedied the trouble.

On 11-7-60 several attempts to start the diesel for the normal weekly test were unsuccessful. A preliminary investigation revealed the safety shut-off control assembly would not reset. A mechanic was called from the Stowers Company, the local Caterpillar distributor, to repair the diesel. A thorough check of the diesel revealed three abnormal conditions: (1) a malfunctioning safety shut-off control assembly; (2) a leak in the fuel supply pipe at a flare fitting between the rigid and flexible pipe; and (3) a faulty cell in the battery.

All three items were replaced by new units after which the diesel operated satisfactorily. Cause of the failure of the shut-off control assembly is not known. This is the second time this unit has been replaced. The leak in the fuel supply line probably resulted from someone having stepped on the line in climbing on the engine. The cause of the battery failure is not known; however, the excessive drain on the battery during the many attempts to start the diesel may have contributed to the failure. While the diesel was being repaired, emergency power was supplied to the scrubber by a temporary line which was run from a small gasoline-powered motor generator set located at the LITR.

On 11-21-60 the diesel engine was slow to start, requiring approximately 25-30 seconds rather than the normal 5-10 seconds. Investigation revealed that the thermostat on the water preheaters was set on 60°F instead of the normal setting of 150°F. The thermostat was adjusted to 150°F and secured in this position so that no one could inadvertently change the setting.

On 11-25-60 after the diesel generator had been operating for about 1 1/2 hr during the regular four-week load test, the building air-conditioning compressor started causing the building lights to dim and dropping the diesel rpm to 900 from the normal rpm of 1200. The air-conditioning compressor and about 50 amps of building lighting load were dropped from the diesel load, reducing the total load on the diesel to about 200 amps. It was, however, impossible to increase the rpm of the diesel to normal even with the frequency adjustment. The diesel manifolds were found to be red hot after the air-conditioning compressor had operated about 5-10 minutes. On reducing the diesel load, the manifolds cooled off. However, when the building load was transferred back to TVA and the diesel operated with essentially no load, raw fuel oil was observed leaking from the manifolds. Replacement of the fuel injectors apparently has corrected the trouble.

On four occasions during power failures, the diesel started and assumed the load as designed. The reliability of this unit still remains in question in view of the failures detected during routine checking.

Gasoline-Driven Pump

The gasoline-driven pump delivers approximately 1000 gpm to provide emergency cooling for fission product afterheat. A routine check for operability has been made once per day by manually starting the unit. This routine check resulted in five failures out of approximately 120 tests during the past six months. On two other occasions the motor stopped after running for extended periods of time. The predominant causes of failure have been attributed to malfunctioning of the automatic choke and the ignition system.

It has been decided to revise the testing procedure to operate the engine on a weekly schedule and run it for one hour thereby permitting the unit to reach operating temperature. It should be pointed out that present conditions do not permit testing the unit with load; however, a study is being made to apply a load to the engine under test which should minimize the fouling of the ignition system.

To date, emergency conditions requiring the services of the unit have not arisen.

D.C. Units No. 1 and No. 2

Concurrent with 30-Mw operation, battery-driven pony motors directly coupled to the main reactor circulating pumps were installed on No. 1 and No. 2 motors. These units are designed to provide a minimum of 500 gpm reactor water flow as a last line of defense in the event of power failure to the main pumps. Actual tests indicate each will produce in excess of 1000 gpm for more than four hours after failure of the main pump power.

The circuitry associated with these D.C. motors includes a bank of eighteen battery cells which are charged by an A.C. supplied battery charger. The present monitoring system is limited to indicate the positions of disconnects located in the battery-to-motor and charger-to-battery circuits. A more detailed monitoring system to indicate the condition of the motor, batteries, and charger is being designed and will be installed as soon as possible.

These units have failed several times due to undersized chargers. On three occasions a fuse in the No. 1 unit battery charger has blown, and on two of these occasions the batteries of this unit were found to be below the minimum requirements. On each occasion, the failure was detected during the routine checking of the specific gravity of the cells.

In an effort to provide more reliability, the following steps are being taken:

1. Provide an adequate monitoring system.
2. Provide larger sized battery chargers or otherwise safeguard chargers from failures.
3. Revise circuitry to prevent overloading the battery chargers.

It has been established that a minumum of 500 gpm flow through the reactor is essential to prevent nucleate boiling following shutdown from 30-Mw operation.[1]

Four independent power sources are now available to provide emergency cooling for the reactor. A decision is needed specifying the minimum number of operable units required to remain within the safety requirements.

Decontamination Scrubber

During the past twelve months, the ORR decontamination scrubber has been put through a complete operational eheck prior to the beginning of each operating cycle. The performance tests have revealed no complete failures; i.e., failure to dump the NaOH. However, abnormal conditions have been detected on three occasions: (1) failure of the circulatin g pump due to NaOH solidifying in the housing-impeller region; (2) minor adjustments on the float switches to regulate liquid level in the sump; and (3) sluggish operation of a pneumatic valve requiring disassembly and cleaning. Of the items mentioned, No. 1 is of major significance since without the pump the effectiveness of the scrubber is limited to a single pass volume of the sump. To minimize a failure of this kind, a local start button has been incorporated in the circuit to permit routine checking of the pump without placing the entire unit into operation.

A work order was issued around December 1959 requesting dual-track instrumentation for the scrubber. To date, this is not complete.

Summary

An analysis of performance tests on the ORR emergency equipment is quite revealing. All units, excluding the battery-driven motors directly coupled to the main pumps, depend upon motors, either internal combustion or electric, to start from rest. All units, with the exclusion mentioned, have failed.

We conclude that emergency equipment needed for the ultimate degree of reliability should be continuous-operating type.

Investigation of 40-Mw Operation

A study to determine the feasibility of increasing the power level of the ORR is currently under way. The preliminary investigations have revealed three principal areas in which problems exist. First, there is the question of adequate external heat removal; second, the problem of maintaining safe maximum fuel temperatures; and, finally, the assessment of the consequences of any pressure and temperature stresses

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[1] J. F. Wett, Requirements for Afterheat Removal at 30-Mw Operation for the ORR, ORNL CF-60-6-13, (June 7, 1960).

which may develop as a result of changes required by the increased heat load.

Because of the concern already expressed over the existing temperature and pressure stresses it was decided to investigate the capabilities of the system utilizing the present coolant flow rate (18,000 gpm) and the present exit temperature (135°F). If this is done the temperature differential across the core would be increased from the present 11.5°F by a fraction just equal to the fractional increase in power; i.e., to 15.3°F at 40 Mw. Since the reactor structure temperatures are known to follow very closely the exit water temperature, this is not expected to affect the former to any great extent.

The present cooling tower and heat exchangers are capable of removing 31 Mw of heat at a wet-bulb temperature of 90°F. They can remove 40 Mw at a wet-bulb temperature of 67°F. The wet-bulb temperature rose above this value on about 70% of the days between April 1 and September 30 during the calendar year 1960. On 1% of those days it rose above 90°F, corresponding to 31 Mw; on 8% of the days it rose above 85°F, corresponding to 34 Mw; and on 19% of the days it rose above 80°F, corresponding to 36 Mw.

It seems clear, therefore, that it is not possible to operate during the summer in a steady fashion using the cooling tower only at powers in excess of 30 Mw and still maintain the current exit temperature and flow rate. For this reason an investigation of the use of the Trane coolers connected in parallel with the existing heat exchangers is being made. It appears that it will be possible to remove 40 Mw of heat in this way and still maintain a reactor exit temperature of 135°F even during periods when the wet-bulb temperature reaches 90°F. The flow distribution under these conditions requires that the by-pass be closed thus reducing to some extent the ability to control the coolant temperatures. This effect is under study at the present time.

Heat transfer in the core appears adequate under the foregoing conditions. The maximum nominal surface temperature of the fuel will be 212°F while the maximum hot-channel, hot-spot temperatures is estimated to be 250°F. In the case of a control rod the corresponding values are 230°F and 265°F, respectively. Since the saturation temperatures in the core is 250°F and since boiling does not occur until this is exceeded by at least 20°F (i.e., at 270°F), it appears that local boiling would not occur at 40 Mw. It is expected, however, to give some attention to the causes of these hot-spot factors in order to reduce the maximum surface temperature still farther below boiling.

Two other problems arise once operation in the above manner is contemplated. The first of these is associated with the fact that should a significant fission break occur the Trane coolers become a serious unshielded source of radiation. The magnitude of the source thus developed has been estimated at approximately 400 r/hr one day after a 1% release of fission products and at a point 1 meter from the coolers. A decision should be made by management concerning the advisability of utilizing these coolers.

The second problem is concerned with a startup accident in the case of 40-Mw operation. At this power the level safeties would normally be set to trip at 60 Mw. This is at too high a level to give protection during startup. It will be necessary, therefore, to arrange matters so that these safeties are set at some level below 50 Mw until the reactor has reached an appreciable power level, say 5 or 10 Mw. The level safeties can then be reset to 60 Mw.

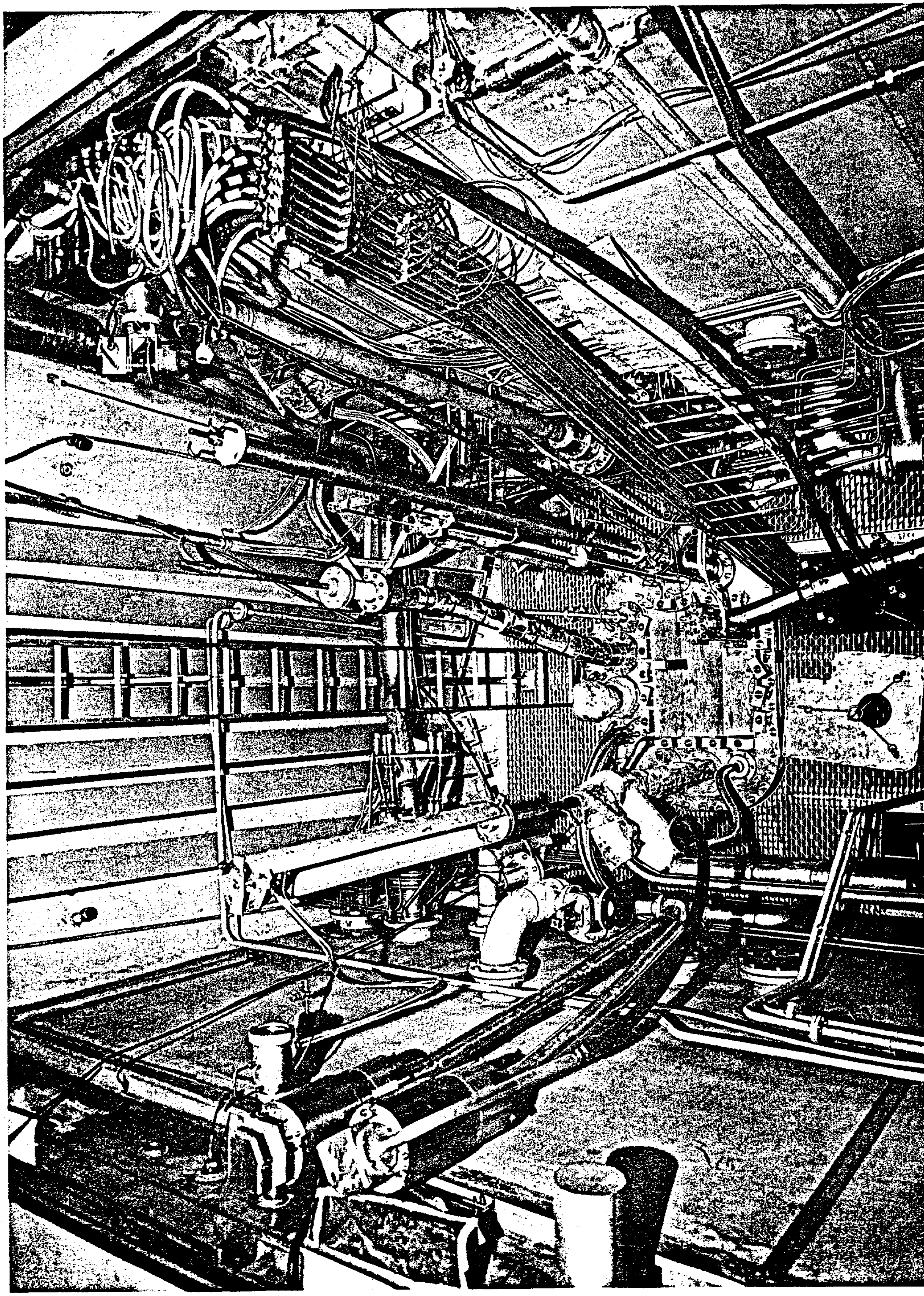
J A Cox

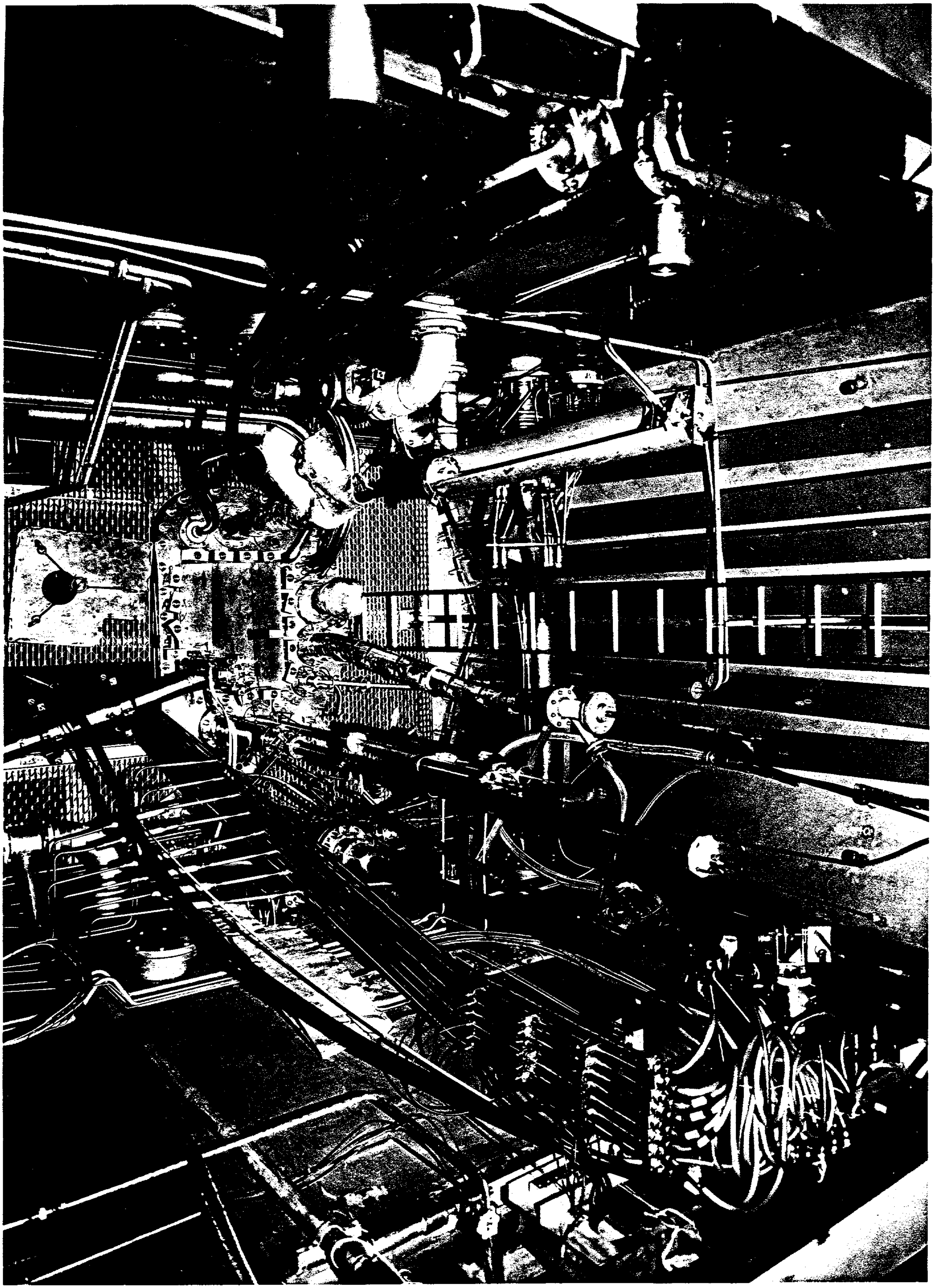
J. A. Cox

JAC:gc

Attachment

cc: F. R. Bruce
A. F. Rupp
A. M. Weinberg





INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

November 20, 1961

To: M. E. Ramsey, D. C. Hamilton, S. H. Hanauer, F. W. Manning, A. M. Perry, R. H. Ritchie, F. Kertesz

Subject: Information for Review of ORR Operation

Prepared By: W. R. Casto; R. A. Costner, Jr.; J. A. Cox; C. D. Cagle; S. S. Hurt, III; L. E. Stanford; W. H. Tabor; and K. W. West

30-Mw Operation

The ORR has operated successfully at a power level of 30 Mw for the past year. Concurrent with 30-Mw operation, an operating cycle of eight weeks was adopted; and this has contributed appreciably to the increased operating time of greater than 80%. An increase in fuel consumption by a factor of about 1.7 has resulted from the increase in power level and the increase in operating time.

Experiments

Pool Experiments

All of the available space for poolside experiments is in use. These experiments do not change appreciably from one cycle to another and, thus, are essentially the same as described in last year's review.

The only notable incident among these experiments occurred during May, 1961, when a plastic tube containing instrument leads and gas-supply lines from the P-6 experiment was accidentally subjected to ~300 psi of helium. The plastic tube ruptured and expelled several gallons of reactor pool water onto the third-floor balcony. The reactor was shut down while the situation was being evaluated; and, after the source of trouble was found, the experiment was deactivated and the reactor returned to power.

A review of the experiment operating procedures and piping layout resulted in several changes in operating technique and piping associated with some of the poolside experiments.

Beam-Hole Experiments

There are five beam-hole experiments operating, and the sixth is scheduled to begin operation in the immediate future.

In-Core Experiments

Seven major experiments utilizing the access flanges through the reactor tank top have operated successfully. Among these is the GCR loop No. 1 experiment, currently operating with the second fuel element. The hydraulic-tube facility No. 1 was recently instrumented to permit the irradiation of fissionable materials. A second hydraulic-tube facility is scheduled for installation during the December shutdown.

Unusual operating conditions experienced with two experiments (B-8, a beryllium tube-burst type; and B-9, a fuel-study type) are included under "Radiation Occurrences."

Engineering-Test Facilities

Work is continuing on the installation of equipment associated with GCR loop No. 2 which will utilize the south engineering-test facility. The containment cell is approximately 99% complete. The 24-inch plug which will contain the loop was installed during the October shutdown. The current schedule indicates the loop will be placed in operation in early 1962.

A fuel-circulating loop is scheduled to begin operating in the HN-1 hole of the north engineering-test facility in early 1962.

Radiation Occurrences

There have been eight "unusual occurrences" at the ORR during the period covered by this report. Six of these resulted in increases in the normal radiation background.

1. On four occasions the B-9 experiment was the contributor to abnormal conditions. The first of these was on May 3, 1961, when water was detected spraying from the B-9 thermocouple header box. The poolside monitron alarmed, indicating a high-radiation level; and air activity in the ORR building approached the alarm point of the constant air monitor. The reactor was immediately shut down and the water spray stopped.

   Investigation indicated that the experiment tube (in which there was no experiment) contained water in a closed system and, due to pressure buildup because of thermal expansion, expelled the contaminated water spray. The experiment valves had not been properly set prior to power operation.

2. Air activity, considerably above normal, existed throughout the ORR building between 6:00 p.m., September 4, 1961, and 9:00 p.m., September 5, 1961, as indicated by constant air monitors. The basement, where the air activity was the highest, was zoned as a contamination zone; and assault masks were worn.

   After an intensive search and study, it was found that the degasifier used on the reactor water system had a faulty condenser which permitted the mixing of the condenser cooling water and the gases removed from the reactor water. This mixture was discharged into the building warm drain sump where the gases separated from the water and contaminated the building air. Removal of the degasifier from operation quickly reduced the air activity to normal.

3. A tube-burst experiment, which was equipped with tantalum thermal shields, was disassembled in the south hot cell on September 22, 1961. During the disassembly the south hot cell was highly contaminated with Ta^{182}.

   During the decontamination of the hot cell, which began on October 10, 1961, the wash was routed to the building hot-drain system. The first wash, a solution of water and detergent, distributed the activity along the hot-drain lines. This created a high-radiation background at points where minimum shielding on the hot-drain line exists. These areas were the first and second levels on the west side of the hot cell and the basement level on the south side of the pool-support structure.

   During a further attempt to reduce the radiation level, a small amount of the decontaminating solution inadvertently escaped into the off-gas system thereby creating an additional radiation problem in the off-gas seal tank located in the subpile room of the ORR basement. Until a formal plan of attack could be made, temporary shielding was placed around the seal tank to reduce the radiation background to a tolerable working level. The Turco decontaminant reduced the radiation levels on the hot-drain lines by as much as a factor of 10; however, temporary shielding was still required to reduce the level of activity. The areas mentioned are zoned as a "radiation zone," and further decontamination work is in progress.

4. The B-9 experiment developed an abnormal condition for the second time on September 26, 1961, when a high background occurred at the reactor poolside due to a leak from the primary system into the purge line which monitors the secondary system of the experiment. The primary source of radiation was the header box which gave a maximum reading of ~1 r/hr. This indicated two abnormal conditions in the experiment: (1) the capsule

was leaking; and (2) a leak existed between the primary and the secondary systems. The capsule was withdrawn from the flux zone, and the experiment was placed in "stand-by" condition for removal during the next scheduled shutdown.

5. On September 26, 1961, the constant air monitors on the second and third levels indicated an increase in background. Investigation revealed the source to be the B-8 beryllium tube-burst experiment. When the Be tube ruptured, radioactive gases (A^{41}) diffused through a leaky back-pressure regulator and into the experiment room atmosphere.

   To prevent a recurrence of this incident, the circuitry was revised to remove the need for the regulator which had malfunctioned.

6. During variable power operation on October 2, 1961, an increase in the radiation level in the B-9 header box at the reactor poolside was again experienced. The maximum reading observed was ~400 mr/hr which decreased rapidly due to air sweep to off-gas and to decay. The fact that radiation was detected only during power variations is attributed to the fact that the pressure in the primary system varies with the temperature; and, during the temperature cycling, the purge-gas system (which operates at a lower negative pressure than the primary system) caused a flow from the contaminated primary system into the purge-gas system. This area was periodically monitored during variable power operation until the capsule was removed during the October shutdown.

7. On October 28, 1961, during the end-of-cycle shutdown, widespread, low-level contamination of the second and third levels of the ORR building occurred while repair work was being performed on the B-9 experiment in-pool piping. A flange was being removed from the in-core experiment tube which had become slightly pressurized due to an undetected air leak. During this removal a cloud of gas escaped and was immediately detected as radioactive by the poolside constant air monitor. The dispersion of the gases and the fallout of the solids produced low-level contamination (maximum 2000 dpm on smears) on the flat surfaces of the second and third levels.

   Clean-up operation was started immediately and all areas cleaned. Personnel in the building at the time of the event were given nasal smears; and only the two persons directly involved with the operation indicated levels above background. "Whole-body" counts were performed on these persons, and initial results indicated <20% of the body burden of the predominant isotope, Cr^{51}. Further analyses are being performed as decay and excretion continues.

8. Contamination occurred on November 3, 1961, during the removal of a vacuum transmitter from the HN-1 equipment chamber at the ORR, the transport of this item to the ORR instrument shop, and thence to Building 3005. The component had not been involved with an operating experiment since September, 1959, and was removed, transported to the ORR instrument shop, disassembled, and taken to the LITR for inspection without a survey by Health Physics. The technician and mechanic involved in the operation chanced to check themselves and found their clothing and shoes to be contaminated. Health Physics was notified, and surveys indicated minor contamination in the ORR instrument shop which was immediately cleaned. All protective clothing worn by personnel involved was discarded. There was no significant radiation exposure to personnel.

Emergency Equipment

The emergency shutdown-cooling system has been improved to provide a high degree of reliability. A number of failures of varying degrees have been experienced on emergency equipment which has been designed to "start-on-demand"; however, the battery-MG set which provides power for the experiments has been reliable in recent tests.

Increased attention to the testing and maintenance programs has improved the reliability of the ORR emergency equipment, particularly in the diesel generator and the battery-operated pony motors. The gasoline-driven emergency cooling pump continued to be troublesome; however, major dependence for shutdown cooling is now placed on the DC pony motors.

Diesel Generator

The diesel generator was tested under load for at least five hours during each of the major reactor shutdowns in addition to the weekly no-load tests of shorter duration. Some improvement in performance is indicated by the fact that the five-hour test runs conducted during the May, July, August, and October shutdowns were satisfactorily completed. Malfunctions which occurred during the January and March test runs are described below.

On January 16, 1961, the test run was stopped after four hours when the diesel generator appeared to be overloaded. A second test was made on January 18, and the trouble was diagnosed as fouled fuel injectors. The injectors were cleaned, and the unit performed satisfactorily.

On March 10, 1961, the automatic speed-regulating mechanism on the diesel generator malfunctioned, preventing the unit from obtaining full speed. Repairs were made and the five-hour test run completed.

One malfunction, discovered during the weekly no-load test on October 16, 1961, was due to leaks in the diesel cooling system. An emergency supply of water was provided, and the leaks were repaired on October 23 during the major reactor shutdown.

The diesel started and operated as designed during five actual power outages.

Gasoline-Driven Pump

Reliance for emergency cooling is no longer placed on the gasoline-driven pump. Instead, the battery-operated motors, which do not have to start from rest, are the main protection from after-heat. During periods when one of the battery-operated motors is not considered reliable, the gasoline pump is operated continuously, or the reactor power is reduced to a level where after-heat is not a problem.

Performance of the automatic starting circuitry for the gasoline pump has been poor. The unit has failed to start on several occasions during routine tests which are conducted during the major reactor shutdowns. Part of the trouble with the circuit resulted from the fact that the operability depended upon the voltage rating of the pump "on" light. This condition has been corrected by the addition of relays to the circuit, and further tests should prove more successful.

Weekly, hour-long, test runs instead of the former, daily, five-minute tests indicate that the performance of the ignition system has been improved.

It is to be noted that emergency conditions requiring the use of this unit have never occurred.

DC Units

The installation of the battery-driven pony motor on the No. 3 main pump was completed July 8, 1961. Since that time, only one occasion has arisen when the batteries of two DC units were concurrently undercharged--a condition which warrants continuous operation of the gasoline pump. Prior to the installation of No. 3 pony motor, it was necessary to operate the gasoline pump continuously on at least six occasions. This trouble is attributed to the undersized battery chargers on units No. 1 and 2. The charger for DC unit 3 is adequate.

Current-limiting relays were installed in the battery-charger circuits for units No. 1 and 2 in December, 1960. These relays have virtually eliminated the blowing of fuses which had occurred on several occasions prior to their installation.

Concurrent with the installation of the No. 3 pony motor, monitors of the condition of the motor, batteries, and charger were added. Current and voltage meters were installed in the reactor control room which permit local read-out; also, low and high alarms were incorporated into the annunciator circuit to provide both audible and visual indications of abnormal conditions of the system.

Decontamination Scrubber and Cell Ventilation

The ORR decontamination scrubber is required to satisfactorily perform a complete operational cycle prior to the beginning of each reactor operating cycle. As reported in the 1960 review, minor component failures still occur; i.e., float switches which regulate the sump level have failed and sluggish operation of a pneumatic valve required disassembly and cleaning.

During the October shutdown, an attempt was made to operate the scrubber unit continuously for four days to determine the reliability of operation. During this period, failures of components were experienced on four occasions. The test was inconclusive except to indicate the present scrubber is unsuited for continuous operation.

A preliminary design and cost estimate is being completed by the Engineering and Mechanical Division to provide an adequate decontamination unit and cell ventilation system which will operate continuously. It is apparent that a "start-on-demand" unit cannot have the reliability of a continuously operating system for emergency usage. Modifications to the present building cell-ventilation system may be necessary to provide extra capacity required by experiment cubicles which rely on this system as a means of secondary containment. Preliminary design includes a charcoal bed to remove the iodine from the ORR building discharge with the major pressure drop of the ventilation system being taken in the experiment cells rather than in the clean-up unit.

Off-Gas

High-Pressure System

The present building off-gas system, which is an original building utility, provides service to both reactor facilities and experiments. A separate off-gas system is now required for all experiments that may discharge radioactive gases under pressure.

A "high-pressure" off-gas system has been designed and installation is in progress. A service manifold to which experiment equipment can be connected will be located in the pools. This system should be ready for operation in early 1962 and will be used where the possibility exists of an experiment pressurizing the off-gas system.

Standard System

Some experiments in the ORR require off-gas service for primary cooling. Should a failure occur in the primary off-gas unit, a criteria for continued reactor operation requires that a "back-up" unit be operative or that the experiment be retracted from the reactor. The units that provide the off-gas service are: (1) an electrically driven compressor, and (2) a steam-turbine-driven compressor. Normal service utilizes the electrically driven unit with the steam unit on standby. If a "back-up" unit is not available for service, all experiments in the category mentioned above are withdrawn from the high-flux zone.

Periodic testing of the "back-up" unit is necessary to give assurance of reliability. The present procedure for testing requires that the steam-driven compressor be operated for about four hours each week with the electrically driven unit shut down. During this testing period, the "back-up" is provided by an operator who is posted at the controls to observe the negative pressure in the off-gas manifold and to restart the electrically driven compressor immediately should any abnormal fluctuations occur.

Cooling System

Reactor Primary System

On April 23, 1961, a series of tests were performed on the ORR primary water system to obtain data on the Trane air coolers operated in parallel with the shell-and-tube heat exchangers. During one phase of the tests, the primary by-pass valve closed although opposed by the pneumatic operator. Two of the primary pumps were stopped when this occurred. A water hammer followed apparently due to the momentary sticking of a check valve. A gasket between the reactor tank top and the access cover was later found to be ruptured, indicating that a shock wave had been transmitted back through the lines to the tank. Precision measurements inside the core housing indicated that no damage had occurred. A detailed examination of other components in the system also revealed no damage.

The primary by-pass valve was equipped with a stop to prevent inadvertent closing. A new valve with a larger operator was installed during October, 1961.

On August 5, 1961, a leak of approximately 15 gpm was observed at the strainer in the 24-inch cooling-water return line. Investigation revealed that the gasket was faulty. Samples of the drainage system to which the water was leaking were taken frequently, and the results indicated that an insignificant increase in activity occurred. The reactor was operated at 30 Mw until 9:00 a.m., August 7, 1961, at which time it was shut down for repair. Refueling was also accomplished during the shutdown which lasted six hours.

Leakage Detection System

Early in this period an increased volume of condensate was collected from the annular region between the concrete and a section of the 24-inch south inlet water line. The section of line involved runs from the pipe chase to the west of a type "I" anchor. A special test was conducted on May 16, 1961, to determine if a leak existed. The water level in the reactor vessel was lowered to expose a portion of the inlet water line, the annular region was pressurized with helium to 20 psi, and a helium leak detector was positioned to monitor the air in the reactor vessel. From analysis of the test results, it was concluded that positive indication of a leak does exist; however, it was impossible to evaluate the exact size of the leak. Routine checks of the volume of condensate collected from this annular region are being continued. Engineering studies have been initiated to evaluate repair methods to be used should a more serious leak occur.

Pool Linér Deformation Study

An engineering study was made to determine the amount of deformation which might be expected should a rupture occur in the reactor inlet water line which is enclosed in the pool-wall structure. The primary concern is if deformation would develop to the extent that operating experiments would be jeopardized by the expanding pool liner.

Results of the study indicate that the liner will fail if subjected to a pressure differential of ~11 psi which would stress the aluminum straps holding the liner to their yield strength of 5000 psi with a deflection of 0.9 inches.

For a small leak which might occur, it is felt that the vacuum system will relieve pressure buildup; and if a sudden rupture were to occur the reactor would be scrammed due to a decrease in the ΔP across the reactor.

The probability of a rupture of a water line causing pool liner deformation which, in turn, might cause a malfunction of an experiment is yet to be determined.

Reactor Secondary System

On August 3, 1961, water was observed leaking at a rate of approximately 200 gpm from a mechanical slip joint on the secondary-pump discharge header. Investigation revealed that a pipe elbow had shifted and that damage also had occurred to a concrete wall through which the pipe passed. It was surmised that water hammer had caused the damage. The pipe was banded and the joints reinforced with tie-rods.

Revisions were made to the secondary fan-control instrumentation to provide better control and to prevent cycling of the fans. Prior to the change, the fan speed was controlled as a function of the basin temperature by four fixed-temperature switches. It is now also controlled as a function of the secondary valve position and, therefore, responds to temperature changes in the system which are quite rapid in comparison to changes in the basin temperature which acts as a heat sink.

Pool Primary System

The pool shell-and-tube heat exchanger was dismantled on May 8, 1961, for inspection and cleaning. Poor heat transfer had been indicated for some time, and an accumulation of silt on the secondary side was suspected. The inspection revealed, however, that the primary side (inside of the tubes) was coated with a brown sludge which was identified as biological in nature. Only a small percentage of the organisms were living, and further checks indicated that growth was probably occurring in the anion section of the pool demineralizer.

Both the secondary and primary sides of the heat exchanger were cleaned. The primary, or tube, side was cleaned by flushing with process water, loosening the foreign matter with long-handled brushes, and reflushing each tube. The secondary side was isolated from the pool cooling system and was cleaned by recirculating a solution of chromic acid, phosphoric acid, and water through this section. Temporary piping was arranged to permit usage of the pool cooling pump for recirculation.

During the July, 1961, shutdown the anion resin of the pool demineralizer was sterilized with a treatment of aqueous solution of a bactericide. Periodic treatments at about six-month intervals are recommended to retard bacteria growth.

The results of the cleaning operation were successful as indicated by the performance of the heat exchanger during subsequent operation. The heat transfer coefficient of the heat exchanger increased to approximately the normal value.

Pool Secondary System

On September 30, 1961, at about 5:30 p.m., a sharp increase in pool secondary water flow was noted. An increase in temperature also occurred with no apparent cause. Inspection of the tower revealed that three of six blades had been broken from the tower fan and that the shaft connecting the motor to the gear box was broken. The reactor power was reduced to 15 Mw until emergency measures could be taken to provide adequate cooling. This was accomplished by adding sufficient cold process water to the system while removing part of the water returning to the tower. The reactor power was increased to 30 Mw after these steps were taken, and repairs of the tower were in progress at the end of the quarter.

It is believed that deflection of the siding on the tower throat caused binding on the fan blade tips and, due to the resulting back torque, the motor-gear reducer coupling broke loose. The fan blades were probably broken as a result of striking this coupling.

Mechanical Controls

The troubles encountered with the ORR mechanical-control mechanism may be categorized as affecting: (1) the shim-rod drive units; (2) the two Cd-Be auxiliary control rods located in core positions F-4 and F-6 and identified as rods No. 2 and 1, respectively; and (3) the hold-down arms. The drive units have on occasion resulted in the shim rods failing to drop, dropping due to unknown cause, and dropping due to magnet failure. The Cd-Be auxiliary control rods "failed to seat" under zero water flow. The hold-down arms do not provide a mechanism for positive latching.

Since November, 1960, no further difficulty has been encountered with the hold-down arms due to a new procedure for checking the units. A Reactor Division "task force" has designed and fabricated a prototype "positive-lock" attachment to the standard hold-down arms. Further study of this design is being made by the "task force", and the prototype has not been placed in service. The difficulties with the Cd-Be rods are also to be studied by this Reactor Division group. Personnel from this group have been assigned to: (1) investigate shim-rod unit failures; (2) provide technical assistance for porper maintenance and shop fabrication work on drive components; (3) provide technical inspections of all drive units during assembly and disassembly; and (4) establish a development program which will provide a trouble-free drive unit and test its components in a hydraulic test stand and under simulated operating flow conditions.

Until a new design has been proved acceptable, the interim corrective program consists of replacing all drive units with units containing: (1) plungers of known hardness (Rockwell C-56 or greater); (2) waterproof magnet coils; and (3) modified magnet keepers which prevent accumulation of water in this area in the event of leakage. Rigid inspections of all assembled drive units will be made before installation.

The following table indicates maintenance which has been required by the drive units.

Table 1. Reactor Mechanical Controls Maintenance

Date	Drive No.	Reason for Maintenance	Action Taken
1-9-61	4	Faulty Magnet.	Unit replaced.
1-20-61	6	Failure to drop with water flow at startup check.	Unit replaced.
1-20-61	1 & 2	Replaced for inspection.	
2-12-61	6	Failure to drop at shutdown check.	Unit replaced with new head, new balls, and new plunger of modified design.
2-13-61	4	Failure to drop on startup check.	Unit replaced with a modified plunger and old head.
2-13-61	3	Wet magnet	Unit replaced.

Table 1. (Continued)

Date	Drive No.	Reason for Maintenance	Action Taken
2-26-61	6	Unit would not release rod during startup check.	Unit replaced with an old unit; unit removed had plunger dimensionally incorrect.
3-12-61	5	Magnet faulty.	Unit replaced with new components.
3-12-61	6	Replaced for inspection.	Unit replaced with new components.
3-24-61	4	Would not recock; found indentations on plunger.	Unit replaced with new components.
4-23-61	1	Failure to drop at shutdown check.	Unit replaced with new components.
4-25-61	2	Failure to drop with water flow during startup checks.	Unit replaced with new components.
5-18-61	5	Shim rod was sticking in shock absorber.	Replaced shock absorber; unit removed contained debris including a small piece of welding rod.
6-5-61	6	On 5-25-61, abnormal recocking action observed; tests indicated proper responses on scram, magnet current, etc.	Unit replaced with new components; also replaced gears and bearings in gear box.
6-24-61	6	Magnet damaged by water leaks.	Unit replaced with new components.
7-7-61	4	Routine replacement.	Unit replaced with rebuilt unit.
7-31-61	1	Failure to drop at shutdown check.	Unit replaced with rebuilt unit.
8-1-61	3	Replaced for inspection.	
8-2-61	4 & 5	Replaced for inspection.	
8-3-61	2 & 6	Replaced for inspection.	
8-5-61	5	Faulty gear box.	Replaced unit with spare.
8-21-61	5	Leaking bellows assembly.	Unit replaced and installed unit leaked; a second replacement was made successfully.

Table 1. (Continued)

Date	Drive No.	Reason for Maintenance	Action Taken
8-22-61	5	Faulty gear box	Replaced unit with spare
10-6-61	1	Failure to drop at shutdown check	Unit replaced with new components
10-24-61	4	Replaced for inspection	Unit replaced with new components
10-26-61	6	Replaced for inspection	Unit replaced with new components
10-27-61	2	Water leak	Unit replaced with new components

Fuel and Selection of Fuel Loading

Fuel Elements

The three hundred 200-gram fuel elements ordered at a cost of approximately $554 each have been received, and usage was begun in the spring of 1961. The supplier has discontinued the fabrication of elements of this type. An order for modified or half-fuel elements will have to be placed soon with another vendor, and it has been proposed to obtain an option for 200-gram elements when ordering the modified elements.

This proposal would result in the following schedule for 200-gram elements:

October 1, 1961	On hand	27 months' supply
January 1, 1963	On hand	12 months' supply
January 1, 1963	Deliveries begin at 25/month	
January 1, 1964	On hand	24 months' supply
January 1, 1965	On hand	12 months' supply
January 1, 1966	Deliveries begin at 25/month	
January 1, 1967	On hand	24 months' supply
January 1, 1967	Re-bid at this point	

A new storage rack for the ORR vault has been designed, approved by the Laboratory Criticality Committee, fabricated, and installed.

Burnup of 200-gram elements to 109 grams (45.5% burnup) has been achieved during 30-Mw operation, although the average is about 125 grams. The number of fuel elements which burn down to "spent" or shipping weight continues at about 14 per month.

Cd-Fuel Control Rods

Subsequent to the increase in the operating power level to 30 Mw and the operating cycle to two months, the replacement of shim rods has been required at more frequent intervals. Two "spent" shim rods from the lower flux positions (B-4 and B-6) are removed from the reactor at the end of each operating cycle, the two partially burned-down shim rods are moved from the higher to the lower flux positions, and two new shim rods are inserted in the higher flux positions (D-4 and D-6). The "spent" shim rods contain 60 to 70 grams U^{235} as compared to 40 to 50 grams previously. Using this procedure, it has been found that 18 days of 30-Mw operation can be obtained with a startup loading which reaches criticality with the shim rods midway between the seat and upper limit.

Fuel Handling, Storage, and Shipment

A second seven-place shipping container was obtained in June, 1961, and, after testing, was placed in service in July, 1961. This, in part, relieved the rather serious storage problem which existed by the first quarter of 1961. At about the time this container was put into service, the truck used for transporting the containers to Idaho was used for other work which resulted in doubling the period between shipments. In November, 1961, a second truck was placed in service; and two shipments have been made in slightly more than one week. It had been planned to modify the original container for loading in the ORR pool as soon as all elements of this type were shipped from the canal in Building 3001. However, over four container loads are to be transferred from the BSF to the canal for ultimate shipment to Idaho; and this will delay the modification at least two months.

Fuel Loading Calculations

The program which will permit the fuel loading to be calculated on the ORACLE is in the final stages of development. The basic concepts of this program have been used for about a year in hand calculations which have been very successful. Critical runs are always made to check these hand calculations (such checks will also be made on machine calculations). Results of these runs have indicated satisfactory loadings, and empirical adjustments in loadings have become infrequent. No negative reactivity credit is assumed for the Cd-Be control rods in determining the operating fuel loading due to the difficulties described above. The computer calculations will have the advantage of greater flexibility and will result in a better inventory than can be achieved by hand.

Reactor Controls and Instrumentation

Major Changes

Gamma Chamber Channels. Several changes were made in the gamma-chamber installation[a] during January, 1961. The first involved providing separate power supplies for the two installed chambers and a method of compensating for the increase in chamber current during a reactor operating cycle. Since the initial load resistance for each channel was determined to give a reading of 100 on a 0-150 scale, a switch was installed to insert a shunt reducing the recorder reading about 25%. Thus, when a reading of about 135 is obtained, the use of the sensitivity switch will reduce the reading to about 100. The higher sensitivity position is used during reactor startup.

During May, 1961, the gamma chamber located northeast of the reactor was removed from the reactor; and two gamma chambers were installed in four-inch-diameter housings (Q-1757-3) with revised bonnets. This permitted the gamma chambers to be installed in the standard ORR ionization-chamber positions[b]. One of these chambers, installed in position No. 8, is tied into the safety system while that installed in position No. 1 is monitored by a spare recorder. The two chambers do not include sensitivity switches because previous investigation had shown that in the standard ionization-chamber positions the increase in chamber current during an operating cycle is insignificant.

In July, 1961, the chamber in position No. 1 was temporarily removed to provide space for a neutron-sensitive detector to be used in conjunction with a boiling-detection study. This gamma chamber was reinstalled during the end-of-cycle shutdown in October, 1961, and is now tied into the safety circuit. At the same time, the one remaining vertical chamber was removed.

Fission Chamber Channel. During cycle 31 in December, 1960, and January, 1961, the fission-chamber drive assembly stuck frequently between 20 and 30 in. withdrawn. As the shutdown progressed, it became apparent that the repair would require a more extensive replacement of parts than was possible at that time. Since preparation for repairing the drive was estimated to require about one month, it was decided to install a temporary fission chamber and preamplifier for use with the permanent channel. The installation of a fission chamber in the ORR pool had been previously investigated[1]. A "Procedure for Checkout and Operation of Temporary ORR

(a) The major objectives of the gamma-chamber installation are: (1) to obtain information regarding the performance of the chambers, especially the variation of chamber current during a normal reactor operating cycle; and (2) to provide a slow scram for protection against an accident caused by the flooding of one or more beam holes during operation which would make the neutron-sensitive channels unreliable.

(b) These standard positions are to the east of the reactor tank and are numbered from south to north as No. 1 through No. 8.

1 D. P. Roux and A. L. Colomb, Experimental Determination of an Adequate Fission Chamber Location in the ORR Pool, ORNL CF-60-11-75 (Nov. 4, 1960).

Fission Chamber" was written by members of the Reactor Controls Department, Instrument and Controls Division. This procedure was followed in detail to establish the adequacy of the temporary channel.

Repair of the permanent fission-chamber drive assembly was begun during the shutdown ending cycle 32 (March, 1961). It was planned to retain the temporary installation as a spare which could be used if necessary. During the repair operation, it was found that the new bearing unit was not interchangeable with the installed one; and use of the temporary channel was continued until repair was completed during the shutdown ending cycle 33 (May, 1961). The temporary poolside channel was retained as an installed spare; however, the guide tube containing the chamber for the channel was removed from the north to the south of the poolside facility.

In July, 1961, during wiring checks of the AC power to the subpile-room fission-chamber preamplifier, it was found that the subpile-room voltage regulation transformer was connected in series with the control-room count-rate-channel voltage regulation transformer. The transformer manufacturer's information indicates that such transformers should not be in series since, under certain load conditions, the reflected impedance will result in an extreme leading power factor. The subpile-room transformer was reconnected to a "raw" AC bus.

Two guide tubes for future permanent poolside channels have been installed on the north side; and testing of appropriate chambers and preamplifiers is in progress.

Level Safety Channels. During the shutdown ending cycle 36 in October, 1961, a modified waterproof can containing a standard ORR PCP safety chamber (Q-975) was installed in the No. 1 level safety channel. The modification consists of combining the chamber-connector head and the waterproof-housing connector plate into a single unit. As a result the chamber was about 10 inches from the nose of the standard waterproof can (Q-1656); the can was then reduced by 10 inches by cutting and welding. This change will aid in improving repair and maintenance as well as in eliminating some items which were highly susceptible to damage by gamma radiation. This modification will be incorporated in the remaining two channels during routine replacements.

Shim-Rod Magnets. During the shutdown ending cycle 35, modifications were made to equipment and special maintenance treatment was performed on the No. 4 and 6 shim-rod-drive units to reduce or prevent moisture from reaching and eventually damaging the insulation of the magnet coils. Drain holes were drilled in the magnet armature according to drawing RC-2-10-4A, revision 4. The special maintenance treatment involved the application of a sealing compound to openings at the top of the magnet shell and the cleaning and subsequent application of the sealing compound to the bottom of the magnet coils. Modification of armatures is scheduled to be completed by the end of December, 1961. The special treatment will be discontinued as soon as modified, moisture-resistant coils are available. During the same shutdown modification of the log count-rate meter (Q-1881) was performed to improve the accuracy of the instrument, particularly at low counting rates. This change also provides for greater accuracy in the period read-out. (Subsequent to the period covered by this report, three of the modified, moisture-resistant coils were installed on shim rods No. 2, 4, and 6. A modified armature was also installed on No. 2 at this time.)

<u>Process Instrumentation</u>. During the shutdown ending cycle 32 in March, 1961, a panelboard was installed southeast of the entrance to the pipe chase to centralize all reactor water-system instruments in this area and to provide more efficient maintenance. This board contains the transmitters for reactor ΔP, facility cooling flow, and off-gas pressure, as well as the nitrogen-16 activity recorder.

During the shutdown ending cycle 34, in July, 1961, modification was made to the reactor controls circuitry associated with the "test" mode of operation. This mode originally permitted operation at low levels under conditions of low water flow by by-passing reactor scrams from low flow, reactor differential pressure, outlet temperatures, and differential temperature. Although control features are provided to prevent operation of the reactor at power levels above about two percent of full power, it was determined that the by-passing of scrams from outlet temperatures and differential temperatures was unnecessary and undesirable. The circuitry was modified so that only scrams from low flow and low reactor differential pressure are by-passed in the "test" mode. During the same shutdown, a modification was made to the circuit which provided for a setback upon the stopping of one of the reactor primary cooling pumps. This setback was to be initiated by a decrease of reactor flow to 16,000 gpm and a reactor differential pressure of 20 psi. It was found that under certain conditions stopping of one pump would not result in reduction of these parameters to the setback point. Since such a setback was determined to be desirable, the setpoints were raised to 17,000 gpm and 21.4 psi, respectively. The point at which the setback was to be terminated was also raised from 0.5 N_F (about 15 Mw) to 0.6 N_F (about 18 Mw) since operation with two pumps at 20 Mw was standard procedure prior to August, 1960.

During the shutdown ending cycle 36 in October, 1961, changes were made in the instrumentation which monitors the coolant flow to the large north and south facilities. Previously the north facility flow was indicated on a gauge and the south facility flow on a recorder installed by GCR Loop No. 2 personnel. These flows were regarded as experiment rather than reactor operating parameters; a standard "E" panel circuit was used to tie in the south facility flow monitoring instrumentation to the reactor control system. As a result of this change the cooling flows for both facilities are now regarded as reactor operating parameters, and their monitoring instrumentation is tied directly into the reactor control system. However, the flow monitoring instrumentation for the GCR Loop No. 2 test plug has an independent tie-in by way of an "E" panel circuit.

Minor Changes

A number of changes was made in instrumentation which were minor in that they did not directly affect the nuclear or reactor-primary process instrumentation. Some of these changes were the culmination of extensive investigations by personnel of Instrumentation and Controls Division, Operating Reactors Group.

Reactor Secondary Cooling System. An annunciator has been provided to warn of low water level in the reactor cooling tower basin. The makeup control for this basin was originally a float-operated valve which was found to give erratic, rather than continuous, correction of water level; this valve was replaced by a pneumatically operated valve. Since this last change was made, control has been more satisfactory; however, the new valve has been observed to have one characteristic of the original (i.e., the basin level is dependent upon reactor power level). This characteristic would be unimportant except that performance tests of the tower indicated that a constant, high, basin level was very desirable. It is planned to install a controller which will maintain a selected basin water level within ± 1/4-inch under all operating conditions.

Two changes were made in the cooling system to improve operation during a reactor scram and subsequent rapid startup. The first change involved installing batch controllers which limit the tendency of the reactor primary and secondary by-pass valve position to go beyond the normal control range following a scram. During a subsequent reactor start-up the time response of the by-pass valves is also improved by these controllers. The second change involved modifying the fan-speed control system for the reactor cooling tower. Since the fan speed is controlled by the basin water temperature, excessive fan-speed cycling was experienced following reactor scrams and subsequent startups. This was due to the response of the secondary by-pass valve causing very low water flow over the tower and through the basin. To alleviate such cycling, a temperature limit controller was installed in the fan-speed control system.

Pool Secondary Cooling System. The controls for the pool secondary cooling system were modified considerably. The primary result of this is that the pool cooling-tower fan speed remains constant until the throttling valve position reaches an extreme of its operating range. The fan speed then adjusts as required to allow the throttling valve to be repositioned.

J. A. Cox

J. A. Cox

JAC:gc

Attachments (Tables 2 through 5)
(Memorandum)

cc: F. R. Bruce; L. B. Holland; L. C. Oakes; A. M. Perry; A. F. Rupp; A. M. Weinberg

Table 2. Unscheduled Shutdowns
Due to Failure of Controls

December, 1960, - September, 1961

Date	Duration (hr)	Remarks
12-22-60	0.033	Shim (auxiliary) rod No. 1 dropped, reason unknown.
12-24-60	0.133	Shim rod No. 5 dropped, reason unknown.
1-5-61	0.150	Shim rod No. 4 dropped, reason unknown.
1-5-61	0.233	Shim rod No. 5 dropped, reason unknown.
1-9-61	19.484	Shim rod No. 4 dropped, Water had leaked through a bellows seal and shorted out the magnet. The drive mechanism was replaced and the reactor reloaded.
2-1-61	0.183	Shim rod No. 3 dropped, reason unknown.
2-1-61	0.150	Shim rod No. 3 dropped, reason unknown.
2-2-61	0.167	Shim rod No. 3 dropped, reason unknown.
3-2-61	0.067	Shim rod No. 5 dropped.[a]
3-2-61	0.066	Shim rod No. 5 dropped.[a]
3-9-61	0.983	Shim rod No. 5 dropped seven times.[a]
3-30-61	0.033	Shim (auxiliary) rod No. 1 dropped, reason unknown.
4-18-61	0.067	Shim (auxiliary) rod No. 1 dropped, reason unknown.
5-25-61	0.167	Shim rod No. 6 dropped, reason unknown. However, investigation after an intentional drop one hour later revealed foreign material on the magnet keeper. This material was removed. The rod then recocked correctly.
5-28-61	0.183	Shim rod No. 6 dropped, reason unknown.
5-31-61	0.167	Shim rod No. 6 dropped, reason unknown.
6-23-61	0.183	Shim rod No. 6 dropped.[b]
6-23-61	0.217	Shim rod No. 6 dropped.[b]
6-23-61	0.167	Shim rod No. 6 dropped.[b]

[a] During the end-of-cycle shutdown, which began on 3-10-61, it was found that the No. 5 shim-rod magnet insulation was breaking down.

[b] Following recovery from the last rod drop on this date, preparation was made to replace the drive tube. Investigation revealed a slight leak in the bellows seal which resulted in magnet failure.

Table 2. (Continued)

Date	Duration (hr)	Remarks
8-21-61	16.867	Shim rod No. 5 dropped. The power was held at N_L; however, investigation revealed water on the magnet and magnet keeper. The drive unit was changed due to a faulty bellows seal. The replacement unit also leaked at the bellows seal, and it was necessary to install a third unit.
8-22-61	5.033	Faulty gear box on No. 5 shim-rod drive was replaced.
9-14-61	0.117	Shim (auxiliary) rod No. 1 dropped, reason unknown.
9-30-61	0.133	Shim (auxiliary) rod No. 1 dropped, reason unknown.

Table 3. Unscheduled Shutdowns
Due to Experiments at the ORR

October, 1960 - September, 1961

Date	Duration (hr)	Remarks
10-19-60	0.033	Setback and reverse from MSR experiment. The recorder on thermocouple TR-12 failed intermittently. The standard cell was replaced.
10-24-60	0.150	Setback from core position F-2 experiment. Thermocouple TC-7 on TR-5 (recorder) opened and resulted in upscale "burnout" from the recorder. The thermocouple was replaced with a spare.
12-22-60	0.283	Setback from EGCR capsule No. 8. Recorder TR-81 drove upscale due to a standardization switch malfunction and stuck the recorder switches. The recorder switches were released and the standardization switch adjusted.
12-22-60	0.217	Setback and reverse from core position F-2 experiment. Recorder TR-11 drove upscale due to electronic tube failure. During the startup shim rod No. 4 dropped, reason unknown.
12-24-60	0.083	Setback and reverse from EGCR capsule No. 8. Recorder TR-81 drove upscale due to standardization switch malfunction and the backup reverse recorder switch stuck. The recorder switches were released and standardization switch adjusted.
12-24-60	0.250	Setback from EGCR capsule No. 8. Recorder TR-81 drove upscale again due to further standardization switch malfunction. The switch was replaced.
2-3-61	7.917	Scram from either GCR Loop No. 1 or MSR (both alarmed). This may have resulted from a momentary loss of instrument power; but extensive investigation failed to indicate the primary cause. Refueling was necessary due to xenon poisoning.
4-4-61	0.083	Setback from MSR experiment. Failure of electronic tube in recorder.
5-6-61	0.350	Scram from GCRP capsule No. 3 experiment. A temperature recorder fuse blew; a replacement fuse also blew; and the "instrument power" breaker was tripped which scrammed the reactor.

Table 3. (Continued)

Date	Duration (hr)	Remarks
6-1-61	0.017	Setback from MSR experiment. Failure of a feedback bellows in a ΔP transmitter resulted in a "low mass flow" indication.
6-25-61	0.083	Setback from MSR experiment. Failure of an electronic tube resulted in upscale "burnout" by a recorder.
7-24-61	0.033	Setback from experiment (CP-C1). Failure of a thermocouple resulted in upscale "burnout" action by the temperature recorder.
9-14-61	0.183	Scram from experiment (CP-B8). The pressure monitor for the experiment can was actuated momentarily when the specimen pressure was bled to the off-gas line too rapidly. The can itself was not pressurized.

Table 4. Analysis of Unscheduled Shutdowns
October, 1960, through September, 1961

Description	Number	Down Time (hr)
Real Cause		
Operations	1	0.050
K-25 (associated with power failure)	1	0.200
Subtotal	2	0.250
Instrument Failure		
Operations	0	0
Experiments	12	9.499
Subtotal	12	9.499
Power failure, due to storm	1	0.350
Mechanical Failure		
Rod drops, undetermined reason	14	1.916
Rod drops, leak resulting in magnet failure	5	36.918
Rod drops, magnet failure	9	1.116
Rod drive gear box replacement	1	5.033
Subtotal	29	44.983
Human Error		
Operations	1	0.217
Experiments	1	0.183
Engineering and Mechanical (resulting in power failure)	1	15.650
K-25 (resulting in power failure)	1	0.200
Subtotal	4	16.250
Total	48	71.332

Table 5. Human Errors and Real Causes
Resulting in Shutdowns at ORR

October, 1960, through September, 1961

Date	Duration (hr)	Description
10-13-60	0.217	Human error, Operations. A false, fast-period scram resulted during the repositioning of the No. 1 Log-N chamber when the chamber slipped too close to the reactor.
3-20-61	0.200	Human error (resulting in power failure), K-25. The wrong transfer switch at the K-25 substation was thrown, resulting in a scram.
3-24-61	15.650	Human error (resulting in power failure), Engineering and Mechanical. The wrong switch was thrown during a test of a transfer switch at the substation, resulting in a scram. Power was not restored for about 15 minutes, and refueling was necessary.
4-15-61	0.200	Real Cause; human error (resulting in power failure), K-25. The K-25 substation operator pulled the wrong breaker resulting in a scram. This disconnected the power to the reactor primary and secondary cooling pumps, reactor cooling tower, pool secondary cooling pump, and pool cooling tower. Since the reactor building retained power and the scram was caused by low cooling flow, an operating parameter, this is classified as a real cause.
9-2-61	0.050	Real Cause, Operations. The reactor was operating at reduced power (3 Mw) for a special thermal stress measurement which involved heating of the reactor water system to the exit temperature setback setpoint. The setting of the recorder switches was such that the "back-up reverse" switch was actuated at a lower temperature than the "setback" switch instead of simultaneously.
9-14-61	0.183	Human error, experiments. A scram resulted when the overpressure monitor for the CP-B8 experiment container was momentarily actuated. The specimen pressure was bled to the off-gas line too rapidly, and a solenoid valve was forced open allowing the monitor to sense the overpressure. Since it is doubtful that the container pressure was actually increased a significant amount (the three pressure-relief valves have a relief area larger than the supply line), this is not classified as a real cause.

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SUBJECT: Minutes of Reactor Operations Review Committee Meeting Held on November 28, 1961

TO: Distribution

FROM: Francois Kertesz

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TX-2701 (1-61)

OAK RIDGE NATIONAL LABORATORY
LABORATORY DIRECTOR'S REVIEW COMMITTEES

Committee: Reactor Operations Review Committee

Meeting Date: November 28, 1961 **Code Number:**

Present:

Members	Experimenters or Operators
M. E. Ramsey, Chairman	F. T. Binford
D. C. Hamilton	C. D. Cagle
S. H. Hanauer	W. R. Casto
L. B. Holland	T. E. Cole
F. Kertesz	R. A. Costner
F. W. Manning	J. A. Cox
A. M. Perry	S. J. Ditto
	R. LeGassie
	R. V. McCord
	L. E. Stanford
	R. S. Stone
	W. H. Tabor

Review of the Oak Ridge Research Reactor

Chairman Ramsey repeated his warning that the review will not relieve the operating group of its responsibility. The Committee review is only to look for any points which have been missed and to assist the operators in making any necessary improvements.

Cox stated that the reactor operated about 80% of the time which is considered very good, being superior to the on-stream time of the MTR and ETR. The operations are carried out with a smaller staff and with 1/3 of the cost of either of the above mentioned two reactors. Recent discussions with the supervisory personnel of MTR-ETR indicated that the ORR has more complicated experiments and, in general, appears to be operating at a higher level of reliability. A great share of credit is due to the excellent maintenance provided at ORNL.

Another advantage is that the ORR experiments are reviewed by a separate group which has no responsibility for operating the experiments. At the MTR the project engineering group is responsible for reviewing, designing, installing, and operating the experiments. This is a basic conflict of interest as far as producing an experiment having good reliability and operability. Also, 38% of burnup of fuel has been obtained at the ORR which compares very favorably with the 25% at the MTR.

The magnitude of the maintenance operation is obvious from a recently prepared list of instruments: there are 467 individual reactor and experiment instruments with about 4800 components which must be maintained at the ORR. The above figure refers to safety instrumentation only without including the control instruments.

TX-2703 (1-61)

Committee: Reactor Operations Review Committee
Meeting Date: November 28, 1961
Subject: Review of the Oak Ridge Research Reactor

Hazards Discussed and Safeguards Suggested

The control rods were causing difficulties during the previous year. What is the present situation?

> The situation has been carefully reviewed, and it is concluded that there is no cause for concern. The present status is described in a memorandum by T. E. Cole (see Documents Submitted). The basic reason for the failure of the rods to drop was traced to a tendency of the balls to indent the 6° slope of the plunger giving rise to increased friction. It was attempted to remedy this situation by using harder plungers; however, it was found that the hardness of the Haynes Stellite used was not uniformly satisfactory. The present practice requires checking of the hardness of each plunger prior to its use. Steps have been taken to ensure safe operation of the rods by a detailed preventive maintenance program.
>
> Further improvement of the rod drive is part of the redesign work scheduled for the near future.

During the inspection by the subcommittee it was noticed that the operators are scrupulous in observing the startup channel as required in the startup procedure. It is felt that the neutron instrumentation must be in good order whenever fissionable materials are redeployed in a reactor even though the shutdown mechanism is considered reliable.

> When the shim rods are hot (just after a shutdown) they contribute most of the neutrons (from γ,n) seen by the fission chamber in its location under the core. Fuel changes in the core are not detected by this fission chamber. This situation will be improved when the new fission-chamber positions, under design for several years, are completed.

During the above mentioned startup operation the counting rate decreased by a factor of about 10 as the rods were withdrawn. The fuel changes in the core could not be noticed on this instrument. A committee member thought that one might go close to criticality before the chamber would indicate it. This condition has existed since the reactor was built, and an auxiliary channel giving a proper response is in the pool in stand-by condition. Requests are outstanding, as noted above, to the Instrumentation and Controls Division for new fission chambers which it is hoped will be ready soon.

What multiplication factor is needed to notice the count-rate reliability?

> One person estimated that the reactor might reach a k value of 0.99 or even higher. From observations, however, under the worst conditions (when the shim rods are hot) the fission chamber shows a normal indication with the rods 5 to 6-in. from criticality. This appears to represent a k of 0.95.

It would be desirable to install an audible or visual detector, or possibly both, on the balcony.

> Such an instrument has already been ordered and will be installed soon. It will be taken from the fifth stage of the scaler. As a general rule the audio signal might not be very reliable. The question also arises as to whether such an instrument should be double-channeled and whether operations should be suspended if it fails.

Under what conditions is an operating neutron detecting instrument required when shifting fuel in an operating reactor?

> A reliable fission-chamber channel would be desirable. The present fission-chamber condition has existed at the ORR since it was first built; its backward response is most pronounced with hot shim rods. When the shim rods have decayed it responds in an almost normal fashion. Also fuel changes cannot be seen on instrumentation at either the LITR or OGR. Because the low level instrumentation cannot "see" fuel changes in the reactor, the fuel loading must be controlled by administrative safeguards. Similar conditions exist at many reactors. It is more difficult to provide this instrumentation in high-power reactors as compared to critical experiments or low-power reactors.

Recent changes in the experiment instrumentation checkout included turning all the E-panel switches to "test" position. Following this they have to be moved to normal position in order to make the scrams operative.

> This situation is carefully controlled by administrative supervision and by direct tests.

At the beginning of an operating cycle with no xenon poisoning, the core is reloaded with elements of weights varying from ~130 to 200 grams. The method used to calculate a loading is based upon empirical data and formulae which have been developed during operation. It is desirable to load the reactor as heavily as permitted; i.e., to obtain criticality with the 4 fuel rods withdrawn as near as feasible, but never less than, half-way out and with the 2 Be rods completely withdrawn. This will provide adequate fuel to operate until the "midcycle" shutdown, providing the core is not xenon poisoned due to unscheduled shutdowns of excessive length. Following a core loading, critical tests are run to ensure that the criteria mentioned above is followed.

When the reactor is reloaded during a midcycle shutdown or an emergency reloading, a variable number of elements which contain xenon are left in the core. A fuel loading calculation is made using the method mentioned above; however, the total mass of the core remains approximately the same. Since the relative value of individual core positions is known, the reloading method provides for a definite range of fuel weight in each position. This loading will, depending upon the length of the shutdown, provide a shimrod reading of between 17 and 19 inches at critical, with a minimum rod insertion due to xenon burnup to ~16 inches.

It was noted that the cutie-pie which was used to monitor the radiation level from fuel elements brought to within 10 ft of the surface of the water was not functioning properly. This cutie-pie was used to supplement the permanently installed poolside monitron which functioned properly. It is suggested that such instruments be kept in a special rack provided with a low-level source so that the instruments could be checked each time they are used.

> This suggestion is accepted by the operators.

It was also noted that the Log-N channel was off by an appreciable amount at N_F, but read very close to the true reading at full power level.

> The main purpose of this instrument is to indicate the approximate neutron flux during startup making sure that the position of the Log-N channel is compatible with the servo channel. Power is reduced by experiment setbacks to 3 N_L.

What is the present range of the servo mechanism?

> The N_L power is about 300 kw, N_F is a factor of 100 higher.

It was found that the fuel loading display board was not kept up to date.

> The board is generally posted within a few days. It is not used as the primary record but as a display showing relative positions of fuel, shim rods, and experiments.

It was noted that during the last year's review (See Minutes of Meeting Held December 5, 1960) that the process drawings were not up to date.

> The 30-megawatt heat-exchanger system drawings are now completely up to date. The other drawings were not reviewed. Many of the new work orders have not been transferred onto the prints.

What are the consequences of this lack of accuracy of the prints?

> Only an occasional delay in maintenance, but no safety problems are involved.

Why are the changes not made immediately?

> Lack of manpower prevents giving too much attention to these problems which have no immediate importance for safety although admittedly it is not good operating practice to have blueprints which do not reflect the exact state of the installation.

A year ago it was considered that the process instrumentation was below standard with respect to the instrumentation of the experiments. What is the present status?

The instrumentation has been upgraded considerably in the meantime. The off-gas, ventilation, and radiation-monitoring systems are much better; but the scrubber is not quite as good as desirable. Further improvements are planned and work orders are being issued. The main difficulty is the manpower situation.

The work orders received by the Instrumentation and Controls Division are not sufficient by themselves to make the necessary changes. They sometimes include vague expressions such as "up date the radiation instrumentation for the ORR". It is highly desirable that the operators outline the problem more precisely. Operations personnel felt that there was a misunderstanding of the intent of such work orders and that this could be resolved by personal contact.

The responsibility for the safety monitoring is accepted by the Operations Division; however, the services of instrumentation experts are needed. Any change introduced in the system is toward upgrading the instrumentation of the reactor to the level of the experiment instrumentation.

How often is it found necessary to move the chambers?

The chambers have to be moved two or three times per operational cycle.

During the previous review it was mentioned that occasional difficulties were experienced with oversized fuel elements.

This difficulty does not exist any more. The elements fabricated by outside suppliers are inspected very rigidly. If elements are found to be outside of tolerance limits they are adjusted at the machine shop. Locally fabricated elements are also similarly inspected. The blueprints used for fabricating these elements are completely up to date.

It was noticed that during the removal of the elements they have to be slightly jiggled or forced.

This condition is not as serious as it was a year ago. It has been much improved by inspecting the fuel carefully.

Have the seat switches been improved during the year?

These switches need frequent adjustment. A more rigid program of preventive maintenance is being followed.

Operation of the reactor with the lid off might be dangerous when the water level is low because of possible high radiation.

Special attention is paid to this situation. Several radiation monitors installed above the reactor provide warning if the radiation becomes high due to operation at powers above N_L. Also Reactor Controls Change Memorandum No. 53, issued on November 13, 1961 included a work schedule for the December shutdown which provides a device to give a low-level safety. Only one safety channel, instead of three, is being converted for low-power protection; and this will make this channel nonstandard with respect to the remaining two channels. The problem of providing low-level protection has been under discussion for the last two years between the Instrumentation and Controls and Operations Divisions.

With respect to the containment of the building, it was pointed out that the scrubber does not operate as satisfactorily as desired.

The containment is checked at each major shutdown. The 400 gallons of caustic solution provide scrubber operation for a minimum of 4 to 5 hours.

If the ventilation system continues to operate, the protection against the maximum credible accident does not require the scrubber according to the latest information; although admittedly, from the viewpoint of strict legal obligation, the scrubber is required.

What is the reliability of the caustic scrubber system?

Failures of one kind or another have been observed in about one-fourth of the tests.

Did the system ever fail and dump the solution without the operator's being aware of it?

The operators were always aware of unscheduled dumping.

What is the method of obtaining the required negative pressure inside the building?

When the scrubber is actuated, the containment system closes the large truck doors and all ventilation louvers automatically.

How is this condition initiated?

A manual button or monitoring electrometer is used to place the building into containment condition. A gamma-sensitive chamber is permanently mounted outside of the control room; it probably would be desirable to install a second instrument to monitor the radiation level in the ventilation duct. Gauges measuring the building suction are located at the south and north side of the first level and at the west side on the third level.

What is the response to the initiation of the containment system?

Very good response is obtained. The only problem arises from a personnel door being slightly open. In the best case a pressure difference of a little more than 0.3 in. of water gauge has been obtained. A vacuum of at least 0.25-in. water gauge is required before the system is considered reliable.

How was this figure arrived at?

Originally 0.5-in. water gauge pressure difference under normal atmospheric conditions was specified in order to preserve containment under adverse conditions; i.e., in case of a strong wind outside the building. Later measurements indicated that 0.18-in. water gauge pressure gradient is required and 0.3 was chosen. Before power operation, actual surveys were made from inside the building which was kept at a pressure difference of $\sim$ -0.4-in. water gauge to the outside to see the reversal of the pressure during a small cyclone. At that time it was agreed that the 0.3-in. water gauge of negative pressure is adequate. This figure was slightly reduced to -0.25-in. water gauge because of the difficulty of maintaining 0.3-in. A design of an improved ventilating system to increase the building vacuum, provide better suction for experiment cubicles and cells, and give a more reliable performance than the scrubber is under way. It is thought this will cost about $100,000. This problem was discussed with the Committee at which time it was agreed to improve the containment system.

Could the presence of trucks cause difficulties by preventing closure of the doors?

Trucks are not allowed to stand in the doorway when the reactor is operating except for the few minutes necessary to come in or to go out. A long truck which would prevent closure might present a significant hazard; however, such long trucks are used very rarely -- usually for the transportation of loops. While the trucks go in or out the doors are blocked.

It was pointed out that the 0.3-in. water gauge of negative pressure for the building has never been officially adopted by the AEC-ACRS. What is the present status of improving the containment of the building?

Plans have been prepared for upgrading the system at an estimated cost of about $100,000. The proposed system is needed for three reasons: (1) the experiment cubicles and cells need to be kept at -0.3 in. water gauge with respect to the building; (2) the building vacuum needs to be kept at -0.3-in. water gauge more reliably than can be done at present; and (3) the iodine removal system should be more reliable than the present scrubber. If it is not approved, attempts will be continued to make the present system more reliable.

How long would it take to complete this program?

Depending on the construction schedule it probably would take about 7 months.

The system will be continuously operating. A second electrometer will probably be installed to double track the closure of doors, louvers, etc.

Must the reactor power be reduced when the containment is lost?

Under normal operating conditions a setback on cell ventilation will induce a reduction of power.

The LITR, which operates at a power level of 3 Mw, does not have the required containment. May the ORR operate up to that level without containment features?

Definitely not. Operation of the LITR does not present a precedent for any reactor.

Is there a radiation monitor on the exhaust system?

There is none at present although it appears desirable to place a monitor at that location.

What is the status of the water-activity monitor and of the other indicating devices on the exhaust streams?

The circulating water system is monitored and an ion chamber will be installed on the off-gas from the degasifier.

Why isn't an MTR-type monitoring device used for monitoring fission products in the water?

The reaction time of that instrument is too slow; a faster response is required.

It was reported that a plastic tube containing instrument leads and gas supply lines from the P-6 experiment was accidentally ruptured expelling several gallons of reactor pool water onto the third floor balcony. What steps have been taken to prevent such occurrence?

Actually the rupture of the hose represented a safety relief valve. It became obvious that a better rupture arrangement is needed, and pressure relief valves are required and have been installed. A pressure alarm signal has also been provided.

Could other experiments present the same hazard?

The other pertinent experiments were checked, changes were suggested, and the research people were informed of how to handle this problem.

It was reported that the scrubber dumps the caustic occasionally; however, Operations is aware at the time of such unscheduled dumping. The question is whether steps are taken to refill it before restarting the device?

Yes, the scrubber is always checked after such an event and the tank refilled. It does not have to be restarted. It has to be stopped; i.e., the pump turned off and made ready for starting at some future time if a need arises.

If the scrubber stops operation how long does it take to start it again?

It might take several hours; the components are always checked. Under present conditions this would happen only during a test. No occasion has ever actually required that the scrubber operate.

When the sump is emptied how long does it take to dry it up?

This requires a minimum of 4 hours of operation.

Did the system ever operate dry for 4 hours?

This does not occur except in normal operation when it always operates dry. It is wet only during tests or accidental dumping of the caustic.

It is drained when the sump is emptied at a rate of 30 to 40 gallons per minute by pumping to the waste tank. During a 2- to 4-hour test period a float switch may stick, but in no case was it ever operated dry.

Did the recirculation pump ever fail?

Upon startup, failures were occasionally experienced for a few minutes which meant that there was no spray but the system was still wet.

What plans are made for the off-gas system?

A separate off-gas system for pressurized experiments is already being installed and should be ready soon.

Before such a system can be obtained, does the operation of the experiments present any hazards?

The reactor must be shut down to avoid experiments' sending radioactive gas to other buildings should the present off-gas be lost.

What problems arise in connection with 40-Mw operation?

This whole problem was reviewed once, although no definite action has been taken. It will be discussed at a later time as operation at that level is not imminent due to the question of providing the additional cooling for the reactor and the long lead time required by many experimenters.

TX-2703 (1-61)

Committee: Reactor Operations Review
Meeting Date: November 28, 1961
Subject: Review of the Oak Ridge Research Reactor

Did the recent water-hammer problem cause any difficulties?

A gasket was loosened in the reactor tank; accordingly check valves will be installed. Work orders were issued and are being worked on by the Engineering and Mechanical Division with completion expected in about 2 to 3 months.

What is the emergency power situation with respect to the status last year?

This situation has been greatly improved.

Recommendations:

1. Install a fission chamber in such a location so as not to be sensitive to control rod γ,n's and yet be sensitive to fuel manipulation and n multiplication. It is recognized that this is already being done, but it is believed that the job should receive more priority.

2. It appears desirable to change the operation in such a way as to require a minimum change in the position of the ion chambers. This problem may require a design study.

3. The containment situation is to be further reviewed on the basis of our implied commitment with AEC. All other commitments to AEC are also being reviewed.

Documents Submitted:

1. Memorandum prepared by W. R. Casto; R. A. Costner, Jr.; J. A. Cox; C. D. Cagle; S. S. Hurt, III; L. E. Stanford; W. H. Tabor; and K. W. West entitled "Information for Review of ORR Operation" dated November 20, 1961.

2. ORO Reactor Safeguards Review, Oak Ridge Research Reactor, dated April 28, 1961.

Submitted by Francois Kertesz
Francois Kertesz, Executive Secretary
Laboratory Director's Review Committees

February 13, 1962

FK:bMcH

TX-2705 (1-61)

LABORATORY DIRECTOR'S REVIEW COMMITTEES

Committee: Reactor Operations Review
Meeting Date: November 28, 1961
Subject: Review of the Oak Ridge Research Reactor

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